**Joseph Pacher**
Site Vice President

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December 21, 2011

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

**ATTENTION:** Document Control Desk

**SUBJECT:** **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244, 72-67 (ISFSI)

**Emergency Action Level Changes**

**REFERENCE:**
(a) Nuclear Energy Institute (NEI) 99-01, Revision 5, "Methodology for Development of Emergency Action Levels," dated February 2008 (ADAMS Accession No. ML080450149)

(b) NUMARC/NESP-007, Revision 2, "Methodology for Development of Emergency Action Levels", dated January 1992 (ADAMS Accession No. ML041120174)

R.E. Ginna Nuclear Power Plant requests Nuclear Regulatory Commission (NRC) approval for the adoption of revised Emergency Action Levels (EALs) for use at the R.E. Ginna Nuclear Power Plant in accordance with 10 CFR Part 50, Appendix E, Section IV(B)(1). The revised EALs are based on Reference (a). Current R.E. Ginna EALs are based primarily on reference (b).

NRC approval is requested by December 31, 2012, with the EAL changes being implemented within one year after approval. The implementation of the revised EALs will be based on site activities allowing for emergency responder training and familiarization as part of the implementation. Site activities that must be coordinated with implementation of the revised EALs include scheduled refueling outages, initial licensing examinations and annual examinations of licensed operators.

The proposed EAL schemes were developed using the generic development guidance from NEI 99-01, Revision 5 with differences and deviations based upon design criteria applicable to the site as well as licensee preferences for terminology, format, and other licensee desired modifications to the generic EAL scheme provided in NEI 99-01 Revision 5. The instrumentation used to determine EAL entry criteria was evaluated for appropriateness, ranges of indication, and set points as part of the upgrade project.

AX45
NRR
NMSS

Attachment (1) provides the EAL Technical Bases (strike out version) with Attachment (2) providing the EAL Technical Bases (clean version). The Technical Bases document provides an explanation and rationale for each EAL. Attachment (3) provides the EAL Comparison Matrix, providing a line-by-line comparison of the EALs contained in NEI 99-01, Revision 5 to the proposed R.E. Ginna EALs. Attachment (4) contains the Radiation Monitor Supporting Calculations. Attachment (5) contains the EAL Wallchart.

Should you have questions regarding this matter, please contact Mr. Thomas Harding at (585) 771-5219 or Thomas.HardingJr@cengllc.com.

Very truly yours,

Joseph E. Pacher

STATE OF NEW YORK:
: TO WIT:
COUNTY OF WAYNE:

I, Joseph E. Pacher, being duly sworn, state that I am Vice President, R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this request on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of MONROE, this 21st day of December, 2011.

WITNESS my Hand and Notarial Seal: Sharon L Miller
Notary Public

SHARON L. MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 20 14

My Commission Expires:

12-21-14

Attachments: (1) EAL Technical Bases (strike out version)
(2) EAL Technical Bases
(3) EAL Comparison Matrix
(4) Radiation Monitor Supporting Calculations
(5) EAL Wallchart

cc: W.M. Dean, NRC
D.V. Pickett, NRC
Resident Inspector, NRC (Ginna)

# ATTACHMENT (1)

# EAL TECHNICAL BASES (Strike Out Version)

GINNA STATION

CONTROLLED COPY NUMBER ________

PROCEDURE NO. EPAD X-X REV. No. [Draft H]

# EMERGENCY ACTION LEVEL TECHNICAL BASES DOCUMENT

Coordinator, Emergency Preparedness
RESPONSIBLE MANAGER

xx/xx/xx
EFFECTIVE DATE
[Draft H 12/20/11]

CATEGORY x.x

THIS PROCEDURE CONTAINS 336 PAGES

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**ACRONYMS & ABBREVIATIONS**

AC.......Alternating Current
APRM.......Average Power Range Meter
ATWS.......Anticipated Transient Without Scram
CCW.......Component Cooling Water
CDE.......Committed Dose Equivalent
CE.......Combustion Engineering
CFR.......Code of Federal Regulations
CNMT.......Containment
CSF.......Critical Safety Function
CSFST.......Critical Safety Function Status Tree
DC.......Direct Current
DHR.......Decay Heat Removal
Disch.......Discharge
DOT.......Department of Transportation
EAL.......Emergency Action Level
ECCS.......Emergency Core Cooling System
ECL.......Emergency Classification Level
ED.......Emergency Director
EOF.......Emergency Operations Facility
EOP.......Emergency Operating Procedure
EPA.......Environmental Protection Agency
EPG.......Emergency Procedure Guideline
EPRI.......Electric Power Research Institute
ERG.......Emergency Response Guideline
EPIP.......Emergency Plan Implementing Procedure
ESF.......Engineered Safety Feature
ESW.......Emergency Service Water
FAA.......Federal Aviation Administration
FBI.......Federal Bureau of Investigation
FEMA.......Federal Emergency Management Agency
FSAR.......Final Safety Analysis Report
GE.......General Emergency
HOO.......Headquarters (NRC) Operations Officer
HPSI.......High Pressure Safety Injection
IC.......Initiating Condition
IPEEE.......Individual Plant Examination of External Events (Generic Letter 88-20)

## ACRONYMS & ABBREVIATIONS (continued)

ISFSI ........ Independent Spent Fuel Storage Installation
Keff ........ Effective Neutron Multiplication Factor
LCO ........ Limiting Condition of Operation
LER ........ Licensee Event Report
LOCA ........ Loss of Coolant Accident
LPSI ........ Low Pressure Safety Injection
LWR ........ Light Water Reactor
MSIV ........ Main Steam Isolation Valve
MSL ........ Main Steam Line
mR ........ milliRoentgen
MW ........ Megawatt
MWS ........ Miscellaneous Waste 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
NUMARC ........ Nuclear Management and Resources Council
OBE ........ Operating Basis Earthquake
OCA ........ Owner Controlled Area
ODCM ........ Off-site Dose Calculation Manual
ORO ........ Off-site Response Organization
OTCC ........ Once Through Core Cooling
PA ........ Protected Area
PAG ........ Protective Action Guideline
POAH ........ Point of Adding Heat
PRA/PSA ........ Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR ........ Pressurized Water Reactor
PSIG ........ Pounds per Square Inch Gauge
R ........ Roentgen
RCC ........ Reactor Control Console
RCIC ........ Reactor Core Isolation Cooling
RCS ........ Reactor Coolant System
rem ........ Roentgen Equivalent Man
RETS ........ Radiological Effluent Technical Specifications
RPS ........ Reactor Protection System
RPV ........ Reactor Pressure Vessel

**ACRONYMS & ABBREVIATIONS (continued)**

RVLIS ........ Reactor Vessel Level Indicating System
RWCU ........ Reactor Water Cleanup
SAE ........ Site Area Emergency
SBO ........ Station Blackout
SG ........ Steam Generator
SI ........ Safety Injection
SPDS ........ Safety Parameter Display System
SRO ........ Senior Reactor Operator
SSE ........ Safe Shutdown Earthquake
TEDE ........ Total Effective Dose Equivalent
TOAF ........ Top of Active Fuel
TSC ........ Technical Support Center
UE ........ Unusual Event
WE ........ Westinghouse Electric
WOG ........ Westinghouse Owners Group
WRNGM ........ Wide Range Noble Gas Monitor

## 1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for the R. E. Ginna Nuclear Power Plant (Ginna). It should be used to facilitate review of the Ginna EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EPIP-1-0 "Ginna Station Event Evaluation and Classification" and the Emergency Action Level Matrix may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training, for explaining event classifications to off-site officials, and for facilitating regulatory review and approval of the classification scheme.

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 in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

## 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 Ginna 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) Revision 4 was 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 5 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL FAQs. Using NEI 99-01 Revision 5 Final, February 2008 (ADAMS Accession Number ML080450149), Ginna conducted an EAL implementation upgrade project that produced the EALs discussed herein.

## 2.2 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon 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. "Loss" means the barrier no longer assures containment of radioactive materials; "Potential Loss" 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 zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
C. Containment (CNMT): 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.

## 2.3 Emergency Classification Based on Fission Product Barrier Degradation

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

*Unusual Event:*

*Any loss or any potential loss of Containment*

Alert:

*Any loss or any potential loss of either Fuel Clad or RCS*

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*

## 2.4 EAL Relationship to EOPs and Critical Safety Function Status

Where possible, the EALs have been made consistent with and utilize the conditions defined in the Ginna Emergency Operating Procedure (EOP) network. While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events which indicate reactor plant safety and/or fission product barrier integrity are threatened. When these symptoms are clearly representative of one of the NEI 99-01 Rev. 5 Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the EOPs, classification of emergencies using these EALs is not dependent upon EOP entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

## 2.5 Symptom-Based vs. Event-Based Approach

To the extent possible, the EALs are symptom-based. That is, the action level threshold is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. However, a purely symptom-based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

## 2.6 EAL Organization

The Ginna 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, Refuel 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 of the above three groups, assignment of EALs to categories/subcategories – category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The Ginna EAL categories/subcategories and their relationship to NEI 99-01 Rev. 5 Recognition Categories are listed below.

**EAL Groups, Categories and Subcategories**

| EAL Group/Category | EAL Subcategory |
|---|---|
| <u>Any Operating Mode:</u> | |
| R – Abnormal Rad Release / Rad Effluent | 1 – Offsite Rad Conditions<br>2 – Onsite Rad Conditions & Spent Fuel Events<br>3 – CR/CAS/SAS Rad |
| H – Hazards and Other Conditions Affecting Plant Safety | 1 – Natural or Destructive Phenomena<br>2 – Fire or Explosion<br>3 – Hazardous Gas<br>4 – Security<br>5 – Control Room Evacuation<br>6 – Judgment |
| E – ISFSI | None |
| <u>Hot Conditions:</u> | |
| S – System Malfunction | 1 – Loss of AC Power<br>2 – Loss of DC Power<br>3 – Criticality & RPS Failure<br>4 – Inability to Reach or Maintain Shutdown Conditions<br>5 – Instrumentation<br>6 – Communications<br>7 – Fuel Clad Degradation<br>8 – RCS Leakage |
| F – Fission Product Barrier Degradation | None |
| <u>Cold Conditions:</u> | |
| C – Cold Shutdown / Refueling System Malfunction | 1 – Loss of AC Power<br>2 – Loss of DC Power<br>3 – RCS Level<br>4 – RCS Temperature<br>5 – Communications<br>6 – Inadvertent Criticality |

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 Sections 2.7 & 2.8, and Attachments 1 & 2 of this document for such information.

## 2.7 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, E, C, H, S and F) 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:

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

<u>Initiating Condition (IC)</u>

Site-specific description of the generic IC given in NEI 99-01 Rev. 5.

<u>EAL Identifier (enclosed in rectangle)</u>

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, E, C, H, S or F)
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)

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 Shutdown, 4 - Hot Standby, 5 - Cold Shutdown, 6 - Refuel, D - Defueled, or All. (See Section 2.8 for operating mode definitions)

Basis:

A Generic basis section provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 5. This is followed by a Plant-Specific basis section that provides Ginna-relevant information concerning the EAL. If the EAL wording contains a defined term, the definition of the term is included at the end of the plant-specific basis discussion.

Ginna Basis Reference(s):

Site-specific source documentation from which the EAL is derived

## 2.8 Operating Mode Applicability (Based on Technical Specifications Table 1.1-1)

1 Power Operation

Reactor shutdown margin is less than Technical Specification minimum required ($K_{eff} \geq 0.99$) and greater than 5% rated thermal power (excluding decay heat).

2 <u>Startup</u>

Reactor shutdown margin is less than Technical Specification minimum required ($K_{eff} \geq 0.99$) and less than or equal to 5% rated thermal power (excluding decay heat).

3 <u>Hot Shutdown</u>

Reactor shutdown margin greater than Technical Specification minimum required ($K_{eff} < 0.99$) with coolant temperature (Tavg) greater than or equal to 350°F.

4 <u>Hot Standby</u>

Reactor shutdown margin greater than Technical Specification minimum required ($k_{eff} < 0.99$) with coolant temperature (Tavg) less than 350°F and greater than 200°F (all reactor vessel head closure bolts fully tensioned).

5 <u>Cold Shutdown</u>

Reactor shutdown margin greater than Technical Specification minimum required ($K_{eff} < 0.99$) with coolant temperature (Tavg) less than or equal to 200°F (all reactor vessel head closure bolts fully tensioned).

6 <u>Refuel</u>

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

D <u>Defueled</u>

All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).

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.

## 2.9 Validation of Indications, Reports and Conditions

All EALs and Fission Product Barrier thresholds assume valid indications. All emergency classifications shall be based upon valid indications, reports or conditions. 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.

## 2.10 Planned vs. Unplanned Events

Planned evolutions involve preplanning to address the limitations imposed by the condition, the performance of required surveillance testing, and the implementation of specific controls prior to knowingly entering the condition in accordance with the specific requirements of the site's Technical Specifications. Activities, planned or unplanned, which cause the site to operate beyond what is allowed by the site's Technical Specifications may result in an EAL threshold being met or exceeded. Planned evolutions to test, manipulate, repair or perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and *activation requirements as long as the evolution proceeds as planned and is within* the operational limitations imposed by the specific operating license. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

## 2.11 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined in other situations, further analyses may be necessary (e.g., coolant radiochemistry following an ATWS event, plant structural examination following an earthquake, etc.). Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

Existing guidance for classifying transient events addresses the period of time of event recognition and classification (15 minutes). However, in cases when EAL declaration criteria may be met momentarily during the normal expected response of the plant, declaration requirements should not be considered to be met when the conditions are a part of the designed plant response, or result from appropriate Operator actions.

There may be cases in which a plant condition that exceeded an EAL was not recognized at the time of occurrence but is identified well after the condition has occurred (e.g., as a result of routine log or record review), and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, should be applied.

2.12 Multiple Simultaneous Events and Imminent EAL Thresholds

When multiple simultaneous events occur, the emergency classification level is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency. Further guidance is provided in RIS 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.

Although the majority of the EALs provide very specific thresholds, the Emergency Director (ED) must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the ED, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded. While this is particularly prudent at the higher emergency classes (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

2.13 Emergency Classification Level Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing. A combination approach involving recovery from General Emergencies and some Site Area Emergencies and termination from Unusual Events, Alerts, and certain Site Area Emergencies causing no long term plant damage appears to be the best choice. Downgrading to lower emergency classification levels adds notifications but may have merit under certain circumstances.

# 3.0 REFERENCES

## 3.1 Developmental

3.1.1 NEI 99-01 Rev. 5 Final, Methodology for Development of Emergency Action Levels, February 2008, ADAMS Accession Number ML080450149

3.1.2 NRC Regulatory Issue Summary (RIS) 2003-18, Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels Revision 4, Dated January 2003 (December 12, 2005)

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

## 3.2 Implementing

3.2.1 EPIP-1-0, Ginna Station Event Evaluation and Classification

3.2.2 EAL Comparison Matrix

3.2.3 EAL Matrix

## 3.3 Commitments

None

## 4.0 DEFINITIONS (ref. 3.1.1 except as noted)

**Affecting Safe Shutdown**

Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable hot or cold shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in hot shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is not "affecting safe shutdown."

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in cold shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is "affecting safe shutdown."

**Airliner/Large Aircraft**

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

**Bomb**

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

**Civil Disturbance**

A group of people violently protesting station operations or activities at the site.

**Confinement Boundary**

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

**Containment Closure**

The site-specific procedurally defined actions taken to secure containment ~~(primary or secondary for BWR)~~ and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to Ginna, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in O-2.3.1A, provide a functional barrier to fission product release.

**Explosion**

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

**Extortion**

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

**Faulted**

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

**Fire**

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

**Hostage**

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

**Hostile Action**

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

**Hostile Force**

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

**Imminent**

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

**Intrusion**

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.

**Normal Levels**

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

**Normal Plant Operations**

Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

**Owner Controlled Area**

The site-specific facilities and property outside the the security Protected Area fence.

**Projectile**

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

**Protected Area**

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

**RCS Intact**

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

**Ruptured**

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

**Sabotage**

Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of sabotage until this determination is made by security supervision.

**Safety-Related Structures, Systems and Components** (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.

~~**Significant Transient**~~

~~An unplanned event involving one or more of the following: (1) automatic turbine runback greater than 25% thermal reactor power, (2) electrical load rejection greater than 25% full electrical load, (3) reactor trip, or (4) safety injection activation, or (5) thermal power oscillations greater than 10%~~

Comment [A1]:

**Site Boundary**

The Site Boundary is approximately a 0.3-mile radius around the reactor.

**Strike Action**

Work stoppage within the Protected Area by a body of workers to enforce compliance with demands made on ~~(site specific)~~Ginna. The strike action must threaten to interrupt Normal Plant Operations.

**Unisolable**

A breach or leak that cannot be promptly isolated from the Main Control Board.

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.~~A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.~~

**Valid**

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

**Visible Damage**

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

**Vital Area**

~~Typically a~~Any ~~site specific~~ areas, normally within the Ginna Protected Area, that contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

## 5.0 GINNA-TO-NEI 99-01 EAL CROSS-REFERENCE

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

| GINNA | NEI 99-01 | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| RU1.1 | AU1 | 1, 2 |
| RU1.2 | AU1 | 3 |
| RU2.1 | AU2 | 1 |
| RU2.2 | AU2 | 2 |
| RA1.1 | AA1 | 1 |
| RA1.2 | AA1 | 3 |
| RA2.1 | AA2 | 2 |
| RA2.2 | AA2 | 1 |
| RA3.1 | AA3 | 1 |
| RS1.1 | AS1 | 1 |
| RS1.2 | AS1 | 2 |
| RS1.3 | AS1 | 4 |
| RG1.1 | AG1 | 1 |
| RG1.2 | AG1 | 2 |
| RG1.3 | AG1 | 4 |
| EU1.1 | E-HU1 | 1 |
| CU1.1 | CU3 | 1 |

| GINNA | NEI 99-01 | |
|---|---|---|
| EAL | IC | Example EAL |
| CU2.1 | CU7 | 1 |
| CU3.1 | CU1 | 1 |
| CU3.2 | CU2 | 1 |
| CU3.3 | CU2 | 2 |
| CU4.1 | CU4 | 1 |
| CU4.2 | CU4 | 2 |
| CU5.1 | CU6 | 1, 2 |
| CU6.1 | CU8 | 2 |
| CA1.1 | CA3 | 1 |
| CA3.1 | CA1 | 1, 2 |
| CA4.1 | CA4 | 1, 2 |
| CS3.1 | CS1 | 1 |
| CS3.2 | CS1 | 2 |
| CS3.3 | CS1 | 3 |
| CG3.1 | CG1 | 1 |
| FU1.1 | FU1 | 1 |
| FA1.1 | FA1 | 1 |
| FS1.1 | FS1 | 1 |
| FG1.1 | FG1 | 1 |
| HU1.1 | HU1 | 1 |
| HU1.2 | HU1 | 2 |
| HU1.3 | HU1 | 3 |
| HU1.4 | HU1 | 4 |

| GINNA | NEI 99-01 | |
|---|---|---|
| EAL | IC | Example EAL |
| HU1.5 | HU1 | 5 |
| HU2.1 | HU2 | 1 |
| HU2.2 | HU2 | 2 |
| HU3.1 | HU3 | 1 |
| HU3.2 | HU3 | 2 |
| HU4.1 | HU4 | 1, 2, 3 |
| HU6.1 | HU5 | 1 |
| HA1.1 | HA1 | 1 |
| HA1.2 | HA1 | 2 |
| HA1.3 | HA1 | 3 |
| HA1.4 | HA1 | 4 |
| HA1.5 | HA1 | 6 |
| HA1.6 | HA1 | 5 |
| HA2.1 | HA2 | 1 |
| HA3.1 | HA3 | 1 |
| HA4.1 | HA4 | 1, 2 |
| HA5.1 | HA5 | 1 |
| HA6.1 | HA6 | 1 |
| HS4.1 | HS4 | 1 |
| HS5.1 | HS2 | 1 |
| HS6.1 | HS3 | 1 |
| HG4.1 | HG1 | 1 |
| HG4.2 | HG1 | 2 |

| GINNA | NEI 99-01 | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| HG6.1 | HG2 | 1 |
| SU1.1 | SU1 | 1 |
| SU3.1 | SU8 | 2 |
| SU4.1 | SU2 | 1 |
| SU5.1 | SU3 | 1 |
| SU6.1 | SU6 | 1, 2 |
| SU7.1 | SU4 | 2 |
| SU7.2 | SU4 | 1 |
| SU8.1 | SU5 | 1, 2 |
| SA1.1 | SA5 | 1 |
| SA3.1 | SA2 | 1 |
| SA5.1 | SA4 | 1 |
| SS1.1 | SS1 | 1 |
| SS2.1 | SS3 | 1 |
| SS3.1 | SS2 | 1 |
| SS5.1 | SS6 | 1 |
| SG1.1 | SG1 | 1 |
| SG3.1 | SG2 | 1 |

## 6.0 ATTACHMENTS

6.1 Attachment 1, Emergency Action Level Technical Bases

6.2 Attachment 2, Fission Product Barrier Loss / Potential Loss Matrix and Basis

# ATTACHMENT 1

# EMERGENCY ACTION LEVEL TECHNICAL 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. Offsite Rad Conditions

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. Onsite Rad Conditions & Spent Fuel Events

Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials or those levels which may preclude access to vital plant areas also warrant emergency classification.

3. CR/CAS Rad

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

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **ANY** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer

**EAL:**

**RU1.1 Unusual Event**

**ANY** gaseous or liquid monitor reading > Table R-1 column "UE" for ≥ 60 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| **Gaseous** | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 µCi/cc | 1.8E+1 µCi/cc | 1.8E+0 µCi/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 µCi/cc | 2.1E+0 µCi/cc | 2.1E-1 µCi/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 µCi/cc | 5.7E+1 µCi/cc | 5.7E+0 µCi/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| **Liquid** | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | with no isolation<br>9.2E+4 cpm with no isolation |

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

The Emergency ~~Director~~ Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This ~~IC~~ EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. ~~[These controls are located in the Off-site Dose Calculation Manual (ODCM), and for plants that have not implemented Generic Letter 89-01, in the Radiological Effluent Technical Specifications (RETS).]~~ The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

~~[Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.]~~

The 2x ~~RETS~~ ODCM limit multiples are specified ~~in AU1 and AA1~~ only to distinguish between non-emergency conditions~~, and from each other~~. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

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

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

~~EAL #1~~

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

This EAL is also intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

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

~~EAL #2~~

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

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

~~EAL #3~~

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

~~EALs #4 and #5~~

~~The 0.10 mR/hr value in EAL #4, and the site specific value for EAL #5,is based on a release rate not exceeding 500 mrem per year.~~

~~[As provided in the ODCM / RETS, prorated over 8766 hours, multiplied by two, and rounded. (500 ÷ 8766 × 2 = 0.114).]~~

~~EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints. EALs #4 and #5 are a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency.~~

~~The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #4 or #5 is an indication of an uncontrolled release.~~

Plant-Specific

Monitor indications are derived from release limits determined by the ODCM methodology and specified offsite dose criteria (ref. 1). These values are summarized in Reference 2.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

**Ginna Basis Reference(s):**

1. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
2. CALC-2011-0020, NEI 99-01 Technical Basis for the Ginna Effluent Monitor Emergency Action Levels (EALs)
3. NEI 99-01 AU1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **ANY** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer

**EAL:**

| **RU1.2** | **Unusual Event** |
|---|---|
| Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x P-9 limits for ≥ 60 min. (Note 2) | |

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This ~~IC~~ EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. ~~[These controls are located in the Off-site Dose Calculation Manual (ODCM), and for plants that have not implemented Generic Letter 89-01, in the Radiological Effluent Technical Specifications (RETS).]~~ The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

~~[Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.]~~

The ~~RETS~~ 2x P-9 (ODCM) limit multiples are specified ~~in AU1 and AA1~~ only to distinguish between non-emergency conditions~~, and from each other~~. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

[Releases should not be prorated or averaged. For example, a release exceeding 4x P-9 (ODCM) for 30 minutes does not meet the threshold.]

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

~~EAL #1~~

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

~~This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.~~

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

~~EAL #2~~

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

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

~~EAL #3~~

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

~~EALs #4 and #5~~

~~The 0.10 mR/hr value in EAL #4, and the site specific value for EAL #5, is based on a release rate not exceeding 500 mrem per year.~~

~~[*As provided in the ODCM / RETS, prorated over 8766 hours, multiplied by two, and rounded. (500 ÷ 8766 × 2 = 0.114).*]~~

~~EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints. EALs #4 and #5 are a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency.~~

~~The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #4 or #5 is an indication of an uncontrolled release.~~

Plant-Specific

Offsite Dose Calculation Manual (ODCM) release limits are specified in Technical Procedure P-9 (ref. 1).

Releases *in excess of two times the site ODCM (ref. 2) instantaneous limits that continue* for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the ODCM limit for 30 minutes does not exceed this initiating condition. Further, the ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
3. NEI 99-01 AU1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **ANY** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer

**EAL:**

| | |
|---|---|
| **RA1.1** | **Alert** |
| **ANY** gaseous monitor reading > Table R-1 column "Alert" for ≥ 15 min. (Note 2) | |

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| **Gaseous** | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 µC/cc | 1.8E+1 µC/cc | 1.8E+0 µC/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 µC/cc | 2.1E+0 µC/cc | 2.1E-1 µC/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 µC/cc | 5.7E+1 µC/cc | 5.7E+0 µC/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| **Liquid** | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm with no isolation |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | 9.2E+4 cpm with no isolation |

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This ~~IC~~ EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. ~~[These controls are located in the Off-site Dose Calculation Manual (ODCM), and for plants that have not implemented Generic Letter 89-01, in the Radiological Effluent Technical Specifications (RETS).]~~ The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

~~[Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.]~~

The ~~RETS multiples are~~ value of 1% (10 mrem) of the EPA PAG threshold (in lieu of 200 times the ODCM release rate limit) is specified ~~in AU1 and AA1 only~~ to ~~distinguish between non-emergency conditions~~ provide a realistic escalation path between the Unusual Event and Site Area Emergency classifications for gaseous releases~~, and from each other~~. While ~~these multiples~~ these thresholds obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant.~~, not the magnitude of the associated dose or dose rate.~~

~~[To ensure a realistic near-linear escalation path, a value should be selected roughly half-way between the AU1 value and the value calculated for AS1 value. The value will be based on radiation monitor readings to exceed 200 times the Technical Specification limit and releases are not terminated within 15 minutes. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. This EAL can be determined using this methodology if appropriate.]~~

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

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

~~EAL #1~~

-This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

Plant-Specific

The values shown correspond to a dose of 10 mrem in one hour at the site boundary. Monitor indications are calculated (ref. 2) based on annual average X/Q dispersion factors from the ODCM (ref. ~~2~~1) and a source term representative of accident conditions. For the main steam line monitors (R-31/32), the variability of results based upon the number of ARVs and/or Main Steam Safety Valves precludes the use of any single default value for these monitors. For these cases, adjustments are made for expected flow rates.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

**Ginna Basis Reference(s):**

1. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
2. CALC-2011-0020, NEI 99-01 Technical Basis for the Ginna Effluent Monitor Emergency Action Levels (EALs)
3. NEI 99-01 AA1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **ANY release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer**

**EAL:**

| | |
|---|---|
| **RA1.2** | **Alert** |
| Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x P-9 limits for ≥ 15 min. (Note 2) | |

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This ~~IC~~ EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. ~~[These controls are located in the Off-site Dose Calculation Manual (ODCM), and for plants that have not implemented Generic Letter 89-01, in the Radiological Effluent Technical Specifications (RETS).]~~ The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

~~[Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.]~~

The ~~RETS~~ 200x ODCM limit~~multiples~~ are specified ~~in AU1 and AA1~~ only to distinguish between non-emergency conditions~~, and from each other~~. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

~~[To ensure a realistic near linear escalation path, a value should be selected roughly half way between the AU1 value and the value calculated for AS1 value. The value will be based on radiation monitor readings to exceed 200 times the Technical Specification limit and releases are not terminated within 15 minutes. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. This EAL can be determined using this methodology if appropriate.]~~

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

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

~~EAL #1~~

~~This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.~~

~~EAL #2~~

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

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

~~EAL #3~~

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

~~EALs #4 and #5~~

~~The 10.0 mR/hr value in EAL #4, and the site specific value for EAL #5,is based on a release rate not exceeding 500 mrem per year.~~

~~[As provided in the ODCM / RETS, prorated over 8766 hours, multiplied by 200, and rounded. (500 ÷ 8766 × 200 = 11.4)].~~

~~EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints. EALs #4 and #5 are a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency.~~

~~The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #4 or #5 is an indication of an uncontrolled release.~~

Plant-Specific

Offsite Dose Calculation Manual (ODCM) release limits are specified in Technical Procedure P-9 (ref. 1).

Releases in excess of two hundred times the site ODCM (ref. 2) instantaneous limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential significant degradation in the level of safety. The final integrated dose (which is very low in the Alert emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 15 minutes. Therefore, it is not intended that the release be averaged over 15 minutes. For example, a release of 400 times the ODCM limit for 7.5 minutes does not exceed this initiating condition. Further, the ED should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
3. NEI 99-01 AA1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

**RS1.1 Site Area Emergency**

**ANY** gaseous monitor reading > Table R-1 column "SAE" for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RS1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| **Gaseous** | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 μCi/cc | 1.8E+1 μCi/cc | 1.8E+0 μCi/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 μCi/cc | 2.1E+0 μCi/cc | 2.1E-1 μCi/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 μCi/cc | 5.7E+1 μCi/cc | 5.7E+0 μCi/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| **Liquid** | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm with no isolation |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | 9.2E+4 cpm with no isolation |

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

This ~~IC~~ EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

~~[While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that for the more severe accidents the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.]~~

~~[The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE...." The EPA PAG guidance provides for the use adult thyroid dose conversion factors. However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facility's emergency planning zone.]~~

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

~~EAL #1~~

The site specific monitor list in ~~EAL #1 should~~ Table R-1 includes effluent monitors on all potential release pathways.

~~[The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site specific boundary (or beyond) dose of 100 mrem whole body or 500 mrem thyroid in one hour, whichever is more limiting (as was done for EALs #2 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.]~~

~~[The meteorology used should be the same as those used for determining AU1 and AA1 monitor reading EALs. The same source term (noble gases, particulates, and halogens) may also be used as long as it maintains a realistic and near linear escalation between the EALs for the four classifications. If proper escalations do not result from the use of the same source term, if the calculated values are unrealistically high, or if correlation between the values and dose assessment values does not exist, then consider using an accident source term for AS1 and AG1 calculations.]~~

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

—Plant-Specific

The values shown correspond to a dose of 100 mrem in one hour at the site boundary. Monitor indications are calculated (ref. 2) based on annual average X/Q dispersion factors from the ODCM (ref. ~~2~~1) and a source term representative of accident conditions. For the main steam line monitors (R-31/32), the variability of results based upon the number of ARVs and/or Main Steam Safety Valves precludes the use of any single default value for these monitors. For these cases, adjustments are made for expected flow rates.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

**Ginna Basis Reference(s):**

1. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
2. CALC-2011-0020, NEI 99-01 Technical Basis for the Ginna Effluent Monitor Emergency Action Levels (EALs)
3. NEI 99-01 AS1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**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 | |

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

This ~~IC~~ EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

~~[While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that for the more severe accidents the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.]~~

~~[The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE...." The EPA PAG guidance provides for the use adult thyroid dose conversion factors. However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facility's emergency planning zone.]~~

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

~~EAL #1~~

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

~~[The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site specific boundary (or beyond) dose of 100 mrem whole body or 500 mrem thyroid in one hour, whichever is more limiting (as was done for EALs #2 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.]~~

~~[The meteorology used should be the same as those used for determining AU1 and AA1 monitor reading EALs. The same source term (noble gases, particulates, and halogens) may also be used as long as it maintains a realistic and near linear escalation between the EALs for the four classifications. If proper escalations do not result from the use of the same source term, if the calculated values are unrealistically high, or if correlation between the values and dose assessment values does not exist, then consider using an accident source term for AS1 and AG1 calculations.]~~

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The 100 mRem TEDE dose is set at 10% of the EPA PAG, while the 500 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Dose assessment may be performed by either manual or computer based methods (ref. 1, 2, 3).

Definitions:

**Site Boundary**

The site boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. EPIP-2-18 Control Room Dose Assessment
2. EPIP-2-5 Emergency Dose Projections - Personal Computer Method
3. EPIP-2-4 Emergency Dose Projections - Manual Method
4. NEI 99-01 AS1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

| | |
|---|---|
| **RS1.3** | **Site Area Emergency** |

Field survey results indicate closed window dose rates > 100 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary

**OR**

Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the site boundary

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

This ~~IC~~ EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

~~[While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that for the more severe accidents the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.]~~

~~[The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE...." The EPA PAG guidance provides for the use adult thyroid dose conversion factors. However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facility's emergency planning zone.]~~

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

~~EAL #1~~

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

~~[*The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site specific boundary (or beyond) dose of 100 mrem whole body or 500 mrem thyroid in one hour, whichever is more limiting (as was done for EALs #2 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.*]~~

~~[*The meteorology used should be the same as those used for determining AU1 and AA1 monitor reading EALs. The same source term (noble gases, particulates, and halogens) may also be used as long as it maintains a realistic and near linear escalation between the EALs for the four classifications. If proper escalations do not result from the use of the same source term, if the calculated values are unrealistically high, or if correlation between the values and dose assessment values does not exist, then consider using an accident source term for AS1 and AG1 calculations.*]~~

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

Real time field surveys and sample analysis is performed by offsite field monitoring teams per EPIP-2-12 "Offsite Surveys" (ref. 1) and assessed for radiological dose consequences per EPIP-2-5 " Emergency Dose Projections - Personal Computer Method" (ref. 2).

Definitions:

**Site Boundary**

The site boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. EPIP-2-12 Offsite Surveys
2. EPIP-2-5 Emergency Dose Projections - Personal Computer Method
3. NEI 99-01 AS1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

**RG1.1 General Emergency**

**ANY** gaseous monitor reading > Table R-1 column "GE" for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RG1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| **Gaseous** | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 µC/cc | 1.8E+1 µC/cc | 1.8E+0 µC/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 µC/cc | 2.1E+0 µC/cc | 2.1E-1 µC/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 µC/cc | 5.7E+1 µC/cc | 5.7E+0 µC/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| **Liquid** | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm with no isolation |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | 9.2E+4 cpm with no isolation |

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

This ~~IC~~ EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

~~[While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that for the more severe accidents the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.]~~

~~[The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "…sum of EDE and CEDE…." The EPA PAG guidance provides for the use adult thyroid dose conversion factors. However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facilities emergency planning zone.]~~

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

~~EAL #1~~

The ~~site specific~~ monitor list in ~~EAL #1 should~~Table R-1 includes effluent monitors on all potential release pathways.

~~[The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site specific boundary (or beyond) dose of 1000 mrem whole body or 5000 mrem thyroid in one hour, whichever is more limiting (as was done for EALs #2 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.]~~

~~[The meteorology used should be the same as those used for determining AU1 and AA1 monitor reading EALs. The same source term (noble gases, particulates, and halogens) may also be used as long as it maintains a realistic and near linear escalation between the EALs for the four classifications. If proper escalations do not result from the use of the same source term, if the calculated values are unrealistically high, or if correlation between the values and dose assessment values does not exist, then consider using an accident source term for AS1 and AG1 calculations.]~~

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The values shown correspond to a dose of 1000 mrem in one hour at the site boundary. Monitor indications are calculated (ref. 2) based on annual average X/Q dispersion factors from the ODCM (ref. ~~2~~1) and a source term representative of accident conditions. For the main steam line monitors (R-31/32), the variability of results based upon the number of ARVs and/or Main Steam Safety Valves precludes the use of any single default value for these monitors. For these cases, adjustments are made for expected flow rates.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

**Ginna Basis Reference(s):**

1. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
2. CALC-2011-0020, NEI 99-01 Technical Basis for the Ginna Effluent Monitor Emergency Action Levels (EALs)
3. NEI 99-01 AG1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**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 | |

**Mode Applicability:**

All

**Basis:**

Generic

~~[*Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.*]~~

This ~~IC~~ EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

~~[*While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that for the more severe accidents the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.*]~~

~~[*The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE...." The EPA PAG guidance provides for the use adult thyroid dose conversion factors. However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facilities emergency planning zone.*]~~

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

~~EAL #1~~

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

~~[The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site specific boundary (or beyond) dose of 1000 mrem whole body or 5000 mrem thyroid in one hour, whichever is more limiting (as was done for EALs #2 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.]~~

~~[The meteorology used should be the same as those used for determining AU1 and AA1 monitor reading EALs. The same source term (noble gases, particulates, and halogens) may also be used as long as it maintains a realistic and near linear escalation between the EALs for the four classifications. If proper escalations do not result from the use of the same source term, if the calculated values are unrealistically high, or if correlation between the values and dose assessment values does not exist, then consider using an accident source term for AS1 and AG1 calculations.]~~

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The 1000 mRem TEDE dose is set at 100% of the EPA PAG, while the 5000 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Dose assessment may be performed by either manual or computer based methods (ref. 1, 2, 3).

Definitions:

**Site Boundary**

The site boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. EPIP-2-18 Control Room Dose Assessment
2. EPIP-2-5 Emergency Dose Projections - Personal Computer Method
3. EPIP-2-4 Emergency Dose Projections - Manual Method
4. NEI 99-01 AG1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

| **RG1.3** | **General Emergency** |
|---|---|
| Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary | |
| **OR** | |
| Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the site boundary | |

**Mode Applicability:**

All

**Basis:**

Generic

~~[Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.]~~

This ~~IC~~ EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

~~[While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that for the more severe accidents the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.]~~

~~[The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE...." The EPA PAG guidance provides for the use adult thyroid dose conversion factors. However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facilities emergency planning zone.]~~

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

~~EAL #1~~

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

~~[*The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site specific boundary (or beyond) dose of 1000 mrem whole body or 5000 mrem thyroid in one hour, whichever is more limiting (as was done for EALs #2 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.*~~]

~~[*The meteorology used should be the same as those used for determining AU1 and AA1 monitor reading EALs. The same source term (noble gases, particulates, and halogens) may also be used as long as it maintains a realistic and near linear escalation between the EALs for the four classifications. If proper escalations do not result from the use of the same source term, if the calculated values are unrealistically high, or if correlation between the values and dose assessment values does not exist, then consider using an accident source term for AS1 and AG1 calculations.*~~]

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

Real time field surveys and sample analysis are performed by offsite field monitoring teams per EPIP-2-12 "Offsite Surveys" (ref. 1) and assessed for radiological dose consequences per EPIP-2-5 " Emergency Dose Projections - Personal Computer Method" (ref. 2).

Definitions:

**Site Boundary**

The site boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. EPIP-2-12 Offsite Surveys
2. EPIP-2-5 Emergency Dose Projections - Personal Computer Method

3. NEI 99-01 AG1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Events

**Initiating Condition:** Unplanned rise in plant radiation levels

**EAL:**

**RU2.1 Unusual Event**

Unplanned water level drop in a reactor refueling pathway as indicated by inability to restore and maintain level > SFP low water level alarm setpoint (Note 3)

**AND**

Area radiation monitor reading rise on **EITHER**:

R-2 Containment

**OR**

R-5 Spent Fuel Pool

Note 3: If loss of water level in the refueling pathway occurs while in Mode 5, 6 or D, consider classification under EALs CU3.1, CU3.2 or CU3.3

**Mode Applicability:**

All

**Basis:**

Generic

This ~~IC~~ EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in unplanned increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

~~EAL #1~~

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

~~[*In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via threshold #1 is appropriate given their potential for increased doses to plant staff.*]~~

The refueling pathway is a ~~site-specific~~ combination of cavities, tubes, canals and pools. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

~~[For example, a refueling bridge ARM reading may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, increased radiation monitor indications will need to combined with another indicator (or personnel report) of water loss.]~~

~~[Application of this EAL requires understanding of the actual radiological conditions present in the vicinity of the monitor. Information Notice No. 90-08, "KR-85 Hazards from Decayed Fuel" should be considered in establishing radiation monitor EALs.]~~

For refueling events where the water level drops below the ~~RPV~~Reactor Vessel flange classification would be via EAL CU3.1, CU3.2 or CU3.3~~2~~. This event escalates to an Alert per EAL ~~AA2~~ RA2.1 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-4.

~~EAL #2~~

~~This EAL addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.~~

~~This EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the Threshold. The intent is to identify loss of control of radioactive material in any monitored area.~~

Plant-Specific

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

The SFP is equipped with a level switch (LC-661) that actuates a low level alarm at 20 in. from the top of the SFP (ref. 1). The minimum level per Technical Specifications is 23 feet above the fuel seated in the SFP (ref. 2).

The definition of "... cannot be restored and maintained above..." allows the operator to visually observe the low water level condition, if possible, and to attempt water level restoration instructions as long as water level remains above the top of irradiated fuel.

When the fuel transfer canal is directly connected to the Spent Fuel Pool and refueling cavity, there could exist the possibility of uncovering irradiated fuel in the fuel transfer canal. Therefore, this EAL is applicable to conditions in which irradiated fuel is being transferred to and from the reactor vessel and SFP.

Technical Specifications requires that refueling cavity water level be maintained 23 ft above irradiated fuel seated in the reactor vessel when moving fuel (ref. 3).

Area radiation monitors R-2 and R-5 are located in the proximity of where spent fuel may be located and have been selected to be indicative of a decrease in radiation shielding due to decreasing refueling pathway water level (ref. 4). While a radiation monitor could detect a rise in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not the fuel is uncovered. For example, the reading on an area radiation monitor located on the refueling bridge may rise due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

This event escalates to an Alert if irradiated fuel outside the reactor vessel is uncovered.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. AR-K-29 SFP HI TEMP 115 °F HI-LO LEVEL 20" 12"
2. Technical Specifications Section 3.7.11 Spent Fuel Pool (SFP) Water Level
3. Technical Specifications Section 3.9.6 Refueling Cavity Water Level
4. P-9 Radiation Monitoring System
5. NEI 99-01 AU2

**Category:** R – Radioactivity Release / Area Radiation

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Events

**Initiating Condition:** Unplanned rise in plant radiation levels

**EAL:**

| | |
|---|---|
| **RU2.2** | **Unusual Event** |
| Unplanned area radiation reading increases by a factor of 1,000 over normal levels | |

**Mode Applicability:**

All

**Basis:**

Generic

This ~~IC~~ EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in unplanned increases in radiation dose rates within plant buildings. ~~These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.~~

~~EAL #1~~

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

~~[*In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via threshold #1 is appropriate given their potential for increased doses to plant staff.*]~~

~~The refueling pathway is a site specific combination of cavities, tubes, canals and pools. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.~~

~~[*For example, a refueling bridge ARM reading may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, increased radiation monitor indications will need to combined with another indicator (or personnel report) of water loss.*]~~

~~[*Application of this EAL requires understanding of the actual radiological conditions present in the vicinity of the monitor. Information Notice No. 90-08, "KR-85 Hazards from Decayed Fuel" should be considered in establishing radiation monitor EALs.*]~~

~~For refueling events where the water level drops below the RPV flange classification would be via CU2. This event escalates to an Alert per AA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-4.~~

~~EAL #2~~

This EAL addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the threshold. The intent is to identify loss of control of radioactive material in any monitored area.

Plant-Specific

Assessment of this EAL may be made with survey readings using portable instruments as well as installed radiation monitors.

Definitions:

**Normal Levels**

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

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. NEI 99-01 AU2

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Events

**Initiating Condition:** Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

**EAL:**

**RA2.1 Alert**

Alarm on **ANY** of the following radiation monitors due to damage to irradiated fuel or loss of water level:

- R-12 Containment Vent Noble Gas
- R-14 Plant Vent Noble Gas
- R-2 Containment
- R-5 Spent Fuel Pool

**Mode Applicability:**

All

**Basis:**

Generic

This ~~IC~~ EAL addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

~~[These events escalate from AU2 in that fuel activity has been released, or is anticipated due to fuel heatup. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.]~~

~~EAL #1~~

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

~~[In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via threshold #1 is appropriate given their potential for increased doses to plant staff.]~~

~~EAL #2~~

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

~~[*For example, a refueling bridge ARM reading may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, increased radiation monitor indications will need to combined with another indicator (or personnel report) of water loss.*]~~

~~[*Application of this EAL requires understanding of the actual radiological conditions present in the vicinity of the monitor. Information Notice No. 90-08, "KR-85 Hazards from Decayed Fuel" should be considered in establishing radiation monitor EALs.*]~~

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.1, RG1.2 or RG1.3~~AS1 or AG1~~.

Plant-Specific

This EAL is defined by the specific areas where irradiated fuel is located such as the refueling cavity, reactor vessel, or spent fuel pool.

The bases for the area radiation alarms include a spent fuel handling accident and are, therefore, appropriate for this EAL. Elevated readings on ventilation monitors may also be indication of a radioactivity release from the fuel, confirming that damage has occurred (ref. 1). However, elevated background at the monitor due to water level lowering may mask elevated ventilation exhaust airborne activity and needs to be considered. However, while radiation monitors may detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Interpretation of these EAL thresholds requires some understanding of the actual radiological conditions present in the vicinity of the monitors.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. NEI 99-01 AA2

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Events

**Initiating Condition:** Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

**EAL:**

| **RA2.2** | **Alert** |
|---|---|
| A water level drop in a reactor refueling pathway that will result in irradiated fuel becoming uncovered | |

**Mode Applicability:**

All

**Basis:**

Generic

This ~~IC addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These~~ events represents a loss of control over radioactive material and represents an actual or substantial potential degradation in the level of safety of the plant.

~~*[These events escalate from AU2 in that fuel activity has been released, or is anticipated due to fuel heatup. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.]*~~

~~EAL #1~~

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

~~*[In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via threshold #1 is appropriate given their potential for increased doses to plant staff.]*~~

~~EAL #2~~

~~This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.~~

~~Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to~~

~~water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered.~~

~~While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.~~

~~*[For example, a refueling bridge ARM reading may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, increased radiation monitor indications will need to combined with another indicator (or personnel report) of water loss.]*~~

~~*[Application of this EAL requires understanding of the actual radiological conditions present in the vicinity of the monitor. Information Notice No. 90-08, "KR-85 Hazards from Decayed Fuel" should be considered in establishing radiation monitor EALs.]*~~

Escalation of this emergency classification level, if appropriate, would be based on RAS1.1, RS1.2, RS1.3, ~~or~~ RAG1.1, RG1.2 or RG1.3.

Plant-Specific

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

There is no indirect indication that water level in the spent fuel pool or refueling cavity has dropped to the level of the fuel other than visual observation. Since there is no level indicating system in the fuel transfer canal, visual observation of loss of water level would also be required. If available, video cameras may allow remote observation. Depending on available level indication, the declared threshold may need to be based on indications of makeup rate or lowering in Reactor Coolant Drain Tank (RCDT) level (ref. 1).

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the reactor vessel flange and the top of spent fuel in the SFP. During refueling activities, this maintains sufficient water level in the refueling cavity, fuel transfer canal and SFP. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (ref. 2, 3).

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment.

**Ginna Basis Reference(s):**

1. ER-SFP.1 Loss of Spent Fuel Pool Cooling
2. Technical Specifications Section 3.9.6 Refueling Cavity Water Level
3. Technical Specifications Section 3.7.11 Spent Fuel Pool (SFP) Water Level
4. NEI 99-01 AA2

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 3 – CR/CAS Rad

**Initiating Condition:** Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

**EAL:**

| | |
|---|---|
| **RA3.1** | **Alert** |

Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:

Control Room (R-1)

**OR**

CAS

**Mode Applicability:**

All

**Basis:**

Generic

This ~~IC~~ EAL addresses increased radiation levels that: impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

The cause and/or magnitude of the increase in radiation levels is not a concern of this ~~IC~~EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other ~~IC~~ EAL may be involved.

~~[At multiple unit sites, the EALs could result in declaration of an Alert at one unit due to a radioactivity release or radiation shine resulting from a major accident at the other unit. This is appropriate if the increase impairs operations at the operating unit.~~]

~~[This IC is not meant to apply to increases in the containment dome radiation monitors as these are events which are addressed in the fission product barrier table.~~]

~~[The value of 15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements", provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.~~]

Areas requiring continuous occupancy include the C~~c~~ontrol R~~r~~oom and ~~as appropriate to the site,~~ any other control stations that are staffed continuously, such as the security alarm stations CAS and SAS. ~~[Typically these areas are the Control Room and the Central Alarm Station (CAS).]~~

Plant-Specific

The Control Room and Central Alarm Station (CAS) must be continuously occupied in all plant operating modes at Ginna.

Area radiation monitor (ARM) R-1 detects radiation levels in the vicinity of the main Control Room. This ARM alarms at 2 mR/hr giving personnel sufficient warning of changing levels (ref. 1). There is no area radiation monitoring system at Ginna for the CAS. Abnormal radiation levels may be initially detected by routine radiological surveys.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. NEI 99-01 AA3

**Category E – Independent Spent Fuel Storage Installation (ISFSI)**

EAL Group: Not Applicable (the EAL in this category is applicable independent of plant operating mode)

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. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

A Notification of 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. This includes classification based on a loaded fuel storage cask/canister confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

A hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HA4.1.

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

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**Category:** E – ISFSI

**Subcategory:** Not Applicable

**Initiating Condition:** Damage to a loaded cask confinement boundary

**EAL:**

| | |
|---|---|
| **EU1.1** | **Unusual Event** |
| Damage to a loaded cask confinement boundary | |

**Mode Applicability:**

Not applicable

**Basis:**

Generic

An ~~NO~~UE in this ~~IC~~ EAL is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

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

Plant-Specific

The Ginna ISFSI utilizes the NUHOMS dry spent fuel storage system.

This EAL addresses any condition which indicates a loss of a cask confinement boundary and thus a potential degradation in the level of safety of the ISFSI. The cask confinement boundary is considered the Dry Shielded Canister (DSC).

Definitions:

**Confinement Boundary**

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

**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.

**Ginna Basis Reference(s):**

1. R. E. Ginna ISFSI USAR
2. ~~NEI 99-01 E-HU1~~

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**Category C – Cold Shutdown / Refueling System Malfunction**

EAL Group: Cold Conditions (RCS 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 RCS integrity, containment closure, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refuel, D – Defueled).

The events of this category pertain to the following subcategories:

1. Loss of AC 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 onsite and offsite power sources for the 480V safeguard buses.

2. Loss of 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 the 125 VDC buses.

3. RCS Level

Reactor Vessel or RCS water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity. RCS levels associated with Category C EALs are listed in Table C-5.

4. RCS Temperature

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

5. Communications

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

6. Inadvertent Criticality

Inadvertent criticalities pose potential personnel safety hazards as well as being indicative of losses of reactivity control.

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** AC power capability to 480V safeguards buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 480V safeguards bus power

**EAL:**

| **CU1.1** | **Unusual Event** |
|---|---|
| AC power capability to 480V safeguards buses reduced to a single power source, Table C-1, for ≥ 15 min. (Note 4)<br>**AND**<br>**Any** additional single power source failure will result in a complete loss of all 480V safeguards bus power | |

Note 4: The ED 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.

| Table C-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Safeguard train A, Buses 14 & 18)<br>• EDG 1B (Safeguard train B, Buses 16 & 17) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel, D - Defueled

**Basis:**

Generic

The condition indicated by this ~~IC~~ EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a ~~station blackout~~complete loss of 480V safeguards bus AC power. This condition could occur due to a loss of off-site power with a concurrent failure of ~~all but~~ one emergency generator to supply power to its emergency bus. The subsequent loss of this single power source would escalate the event to an Alert in accordance with EAL CA1.1~~3~~.

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

~~[At multi-unit stations, the EALs should allow credit for operation of installed design features, such as cross-ties or swing diesels, provided that abnormal or emergency operating procedures address their use. However, these stations must also consider the impact of this condition on other shared safety functions in developing the site specific EAL.]~~

~~[Plants that have a proceduralized capability to cross-tie AC power from an off-site power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC.]~~

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., charging pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are specifically included in Table C-1 because service water pump operation is necessary for decay heat removal while in cold conditions.

There are three offsite power sources available to these buses (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and an Unusual Event must be declared.

There are two onsite emergency AC power sources available in the cold modes:

- EDG 1A
- EDG 1B

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to be capable of supplying one or more safety-related buses within 15 minutes, an Unusual Event is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to an Alert under EAL CA1.1.

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. NEI 99-01 CU3

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** Loss of **all** offsite and **all** onsite AC power to 480V safeguards buses for ≥ 15 min.

**EAL:**

**CA1.1 Alert**

Loss of **all** offsite and **all** onsite AC power, Table C-1, to 480V safeguards buses for ≥ 15 min. (Note 4)

Note 4: The ED 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.

| Table C-1 AC Power Sources | |
|---|---|
| Onsite | • EDG 1A (Safeguard train A, Buses 14 & 18)<br>• EDG 1B (Safeguard train B, Buses 16 & 17) |
| Offsite | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel, D - Defueled

**Basis:**

Generic

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink.

The event can be classified as an Alert when in cold shutdown, refueling, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by EALs in Category R~~Abnormal Rad Levels / Radiological Effluent ICs~~.

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

~~[The companion IC is SS1].~~

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., charging pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are specifically included in Table C-1 because service water pump operation is necessary for decay heat removal while in cold conditions.

There are three offsite power sources available to these buses in the cold modes (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and Alert must be declared.

There are two onsite emergency AC power sources available in the cold modes:

- EDG 1A
- EDG 1B

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If all sources fail to be capable of supplying all safety-related buses within 15 minutes, an Alert is declared under this EAL.

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. NEI 99-01 CA3

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – Loss of DC Power

**Initiating Condition:** Loss of **required** DC power for ≥ 15 min.

**EAL:**

| | |
|---|---|
| **CU2.1** | **Unusual Event** |
| < 108 VDC on **required** 125 VDC buses for ≥ 15 min. (Note 4) | |

Note 4: The ED 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.

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

The purpose of this ~~IC~~ EAL~~and its associated EALs~~ is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations.

~~[This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.]~~

~~[Plants will routinely perform maintenance on a Train related basis during shutdown periods The required busses are the minimum allowed by Technical Specifications for the mode of operation.] It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per CA4.~~

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

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

Plant-Specific

The 125 VDC vital system is divided into two independent and isolated channels. Each channel consists of one battery, two battery chargers, one DC bus and one inverter. Each inverter has an associated vital AC distribution panel board. Power to the DC bus, DC unit control panels, and inverters is supplied by the station batteries and/or the battery chargers. Each battery charger is fully rated and can recharge a discharged battery while at the same time supplying the steady state power requirements of the system.

A separate TSC Battery system is designed with an intertie to each of the two main (A and B) distribution panels for use during maintenance and abnormal conditions.

The safety-related station batteries have been sized to carry their expected shutdown loads following a plant trip and loss of offsite power or following a station blackout without battery terminal voltage falling below 108.6 volts for a period of 4 hours (ref. 1).

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

The loss of the TSC Battery does not constitute an entry condition for this EAL.

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS2.1.

**Ginna Basis Reference(s):**

1. UFSAR Section 8.3.2 Direct Current Power Systems
2. O-6.13 Daily Surveillance Log
3. NEI 99-01 CU7

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** RCS leakage

**EAL:**

| **CU3.1** | **Unusual Event** |
|---|---|
| RCS leakage results in the inability to maintain or restore RCS level within the target band established by procedure for ≥ 15 min. (Note 4) | |

Note 4: The ED 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

**Mode Applicability:**

5 - Cold Shutdown

**Basis:**

Generic

This ~~IC~~ EAL is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

Relief valve normal operation should be excluded from this ~~IC~~EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this ~~IC~~ EAL if the relief valve cannot be isolated.

Prolonged loss of RCS inventory may result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL ~~CA4~~CA3.1.

~~[The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In the refueling mode the RCS is not intact and RPV level and inventory are monitored by different means. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available.]~~

Plant-Specific

This EAL is applicable if RCS level cannot be restored and maintained within the prescribed target band specified by procedure.

**Ginna Basis Reference(s):**

2. AP-RCS.1, Reactor Coolant Leak
3. NEI 99-01 CU1

**Category:** C – Cold Shutdown / Refueling System Malfunction
**Subcategory:** 3 – RCS Level
**Initiating Condition:** RCS Leakage
**EAL:**

| | |
|---|---|
| **CU3.2** | **Unusual Event** |

Unplanned RCS level drop below **EITHER** of the following for ≥ 15 min. (Note 4):

Reactor Vessel flange (84 in. on loop level indicators) (when the level band is established above the flange)

**OR**

RCS level target band established by procedure (when the level band is established below the flange)

Note 4: The ED 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

**Mode Applicability:**

6 - Refuel

**Basis:**

Generic

This ~~IC~~ EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water level below the ~~RPV~~ Reactor Vessel flange are carefully planned and procedurally controlled. An unplanned event that results in water level decreasing below the ~~RPV~~ Reactor Vessel flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the ~~RPV~~ Reactor Vessel flange), warrants declaration of a ~~NO~~UE due to the reduced RCS inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL ~~CA4~~CA3.1.

~~[The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and RPV level and inventory are monitored by different means].~~

~~EAL #1~~

This EAL involves a decrease in RCS level below the top of the ~~RPV~~ Reactor Vessel flange that continues for 15 minutes due to an unplanned event. This EAL is not applicable to decreases in flooded reactor cavity level, which is addressed by EAL RU2.1~~AU2 EAL1~~, until such time as the level decreases to the level of the vessel flange.

~~*[For BWRs]* if RPV level continues to decrease and reaches the Low-Low ECCS Actuation Setpoint then escalation to CA1 would be appropriate.~~

~~*[For PWRs]* If RPV level continues to decrease and reaches the Bottom ID of the RCS Loop then escalation to CA1 would be appropriate.~~

~~EAL #2~~

~~This EAL addresses conditions in the refueling mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of RPV level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.~~

~~Escalation to the Alert emergency classification level would be via either CA1 or CA4.~~

Plant-Specific

The Reactor Vessel flange level (uncorrected) is at 84 in. (252' 6" ele.) on Loop A & B Level Indicators (LIT-432A and B) (ref. 1, 2).

This EAL involves a lowering in RCS level below the top of the Reactor Vessel flange, or the inability to maintain water level above the intended level when level is being intentionally maintained below the flange, that continues for fifteen minutes due to an unplanned event. This EAL is not applicable to drops in flooded refueling pool level (covered by lowering spent fuel pool water level in EAL RU2.1) until such time as the level lowers to the level of the vessel flange. If level continues to lower and reaches the bottom of the RCS Hot Leg reference level (0 in. indicated), escalation to the Alert level under EAL CA3.1 would be appropriate. If the level lowering is accompanied by RCS heatup, escalation to the Alert level under EAL CA4.1 may also be appropriate.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refuel mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refuel mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. Reactor Vessel water level is normally monitored in Refuel mode using the following instruments (ref. 3,5,6):

- Loop A Level Indicator LIT-432A
- Loop B Level Indicator LIT-432B
- Loop B Sightglass
- Cavity Water Level

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. RF-601 Fuel Handling Accident Instructions
2. O-2.3 Draining the Reactor Coolant System to < 84" but > 64"
3. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
4. NEI 99-01 CU2
5. O-15.1 Administrative Requirement Checklist for Entry to Mode 6 and Refueling Conditions
6. O-6.13 Daily Surveillance Log

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** RCS Leakage

**EAL:**

| **CU3.3** | **Unusual Event** |
|---|---|
| RCS level **cannot** be monitored with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage | |

| **Table C-2 RCS Leakage Indications** |
|---|
| • Containment Sump A<br>• Containment Sump B<br>• Auxiliary Building Sump Tank<br>• Reactor Coolant Drain Tank (RCDT) |

**Mode Applicability:**

6 - Refuel

**Basis:**

Generic

This ~~IC~~ EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water level below the ~~RPV~~ Reactor Vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the Reactor Vessel~~RPV~~ flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the Reactor Vessel~~RPV~~ flange), warrants declaration of a ~~N~~OUE due to the reduced RCS inventory that is available to keep the core covered.

~~The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.~~

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA3.1 or EAL CA4.1.

~~[The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and RPV level and inventory are monitored by different means].~~

~~EAL #1~~

~~This EAL involves a decrease in RCS level below the top of the RPV flange that continues for 15 minutes due to an UNPLANNED event. This EAL is not applicable to decreases in flooded reactor cavity level, which is addressed by AU2 EAL1, until such time as the level decreases to the level of the vessel flange.~~

~~[For BWRs] if RPV level continues to decrease and reaches the Low-Low ECCS Actuation Setpoint then escalation to CA1 would be appropriate.~~

~~[For PWRs] If RPV level continues to decrease and reaches the Bottom ID of the RCS Loop then escalation to CA1 would be appropriate.~~

~~EAL #2~~

This EAL addresses conditions in the Refuel mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of ~~RPV~~ RCS level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that ~~RPV~~ RCS inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

~~Escalation to the Alert emergency classification level would be via either CA1 or CA4.~~

Plant-Specific

In this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by Containment Sumps, Auxiliary Building Sump or RCDT level changes (ref. 1, 2). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

**Ginna Basis Reference(s):**

1. UFSAR 5.1.3.6 Design Criteria
2. UFSAR 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
3. NEI 99-01 CU2

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** Loss of RCS inventory

**EAL:**

| **CA3.1** **Alert** |
|---|
| Loss of inventory as indicated by RCS water level < 0 in.<br>**OR**<br>RCS level **cannot** be monitored for ≥ 15 min. with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage (Note 4) |

Note 4: The ED 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.

| **Table C-2 RCS Leakage Indications** |
|---|
| • Containment Sump A |
| • Containment Sump B |
| • Auxiliary Building Sump Tank |
| • Reactor Coolant Drain Tank (RCDT) |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

~~These~~ This EAL~~s~~ serve~~s~~ as a precursor~~s~~ to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further ~~RPV~~ RCS level decrease and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert.

~~EAL #1~~

~~[The BWR Low-Low ECCS Actuation Setpoint/Level 2 was chosen because it is a standard setpoint at which some available injection systems automatically start. The PWR Bottom ID of the RCS Loop Setpoint was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The Bottom ID of the RCS Loop Setpoint should be the level equal to the bottom of the RPV loop penetration (not the low point of the loop).]~~

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

~~EAL #2~~

~~*[In the cold shutdown mode, normal RCS level and RPV level instrumentation systems will usually be available. In the refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.]*~~

~~*[The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency EAL duration. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CG1 basis. Therefore this EAL meets the definition for an Alert.]*~~

If ~~RPV~~ RCS level continues to lower then escalation to Site Area Emergency will be via EAL CS~~2~~3.1, EAL CS3.2 or EAL CS3.3.

Plant-Specific

When RCS water level lowers to 0 in. (uncorrected) on loop level indicators, the bottom of the RCS hot leg level instrument tap is uncovered (ref. 1, 2). This level can be monitored by:

- Loop A Level Indicator LIT-432A
- Loop B Level Indicator LIT-432B
- Loop B Sightglass

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel level lowering and potential core uncovery. The bottom of the hot leg is the level equal to the bottom of the Reactor Vessel loop penetration, not the low point of the loop. This level was chosen because remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The inability to restore and maintain level after reaching this setpoint implies a failure of the RCS barrier.

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

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refuel mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refuel mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed

(Cavity level monitoring with the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted (ref. 7,8).

In the second condition of this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by Containment Sumps, Auxiliary Building Sump or RCDT level changes (ref. 1, 2, 3, 5). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency EAL duration. The interval allows this EAL to be an effective precursor to the Site Area Emergency EAL CS3.1. Therefore this EAL meets the definition for an Alert emergency.

**Ginna Basis Reference(s):**

1. RF-601 Fuel Handling Accident Instructions
2. O-2.3 Draining the Reactor Coolant System to < 84" but > 64"
3. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
4. UFSAR 5.1.3.6 Monitoring Reactor Coolant Leakage
5. UFSAR 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
6. NEI 99-01 CA1
7. O-15.1 Administrative Requirement Checklist for Entry to Mode 6 and Refueling Conditions
8. O-6.13 Daily Surveillance Log

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

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

**EAL:**

**CS3.1 Site Area Emergency**

RCS level **cannot** be monitored with a loss of RCS inventory as indicated by **ANY** of the following for ≥ 30 min. (Note 4):

- Containment radiation R-29 or R-30 > 1.0E+02 R/hr
- Erratic Source Range Nuclear Instrumentation indication
- Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage

Note 4: The ED 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.

| **Table C-2 RCS Leakage Indications** |
|---|
| • Containment Sump A<br>• Containment Sump B<br>• Auxiliary Building Sump Tank<br>• Reactor Coolant Drain Tank (RCDT) |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

Under the conditions specified by this ~~IC~~EAL, continued decrease in RCS~~/RPV~~ level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the ~~RPV~~RCS. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG~~2~~3.1, RG1.1, RG1.2 or R~~1 or~~ AG1.3.

~~EAL #1~~

~~*[6" below the bottom ID of the RCS Loop should be the level equal to 6" below the bottom of the RPV loop penetration (not the low point of the loop). PWRs unable to measure this level should choose the first observable point below the bottom ID of the loop as the EAL value. If a water level*~~

~~instrument is not available such that the PWR EAL value cannot be determined, then EAL 3 should be used to determine if the IC has been met.~~]

~~[Since BWRs have RCS penetrations below the EAL value, continued level decrease may be indicative of pressure boundary leakage.~~]

~~EAL #3~~

~~[In the cold shutdown mode, normal RCS level and RPV level instrumentation systems will usually be available. In the refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.~~]

The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

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

~~This EAL should conservatively estimate a site specific dose rate setpoint indicative of core uncovery (i.e., level at TOAF). For BWRs that do not have installed radiation monitors capable of indicating core uncovery, alternate site specific level indications of core uncovery should be used.~~]

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

Plant-Specific

In Refuel or Cold Shutdown mode, normal RCS level indication may be unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- As water level in the Reactor Vessel lowers, the dose rate above the core will rise. Containment radiation is indicated on R-29 and R-30. The dose rate due to this core shine should result in on-scale Containment radiation monitor indication and possible alarm. Assuming total draindown of the upper cavity, line-of-sight dose rates from a fully exposed upper internal package would be approximately 300 R/hr.

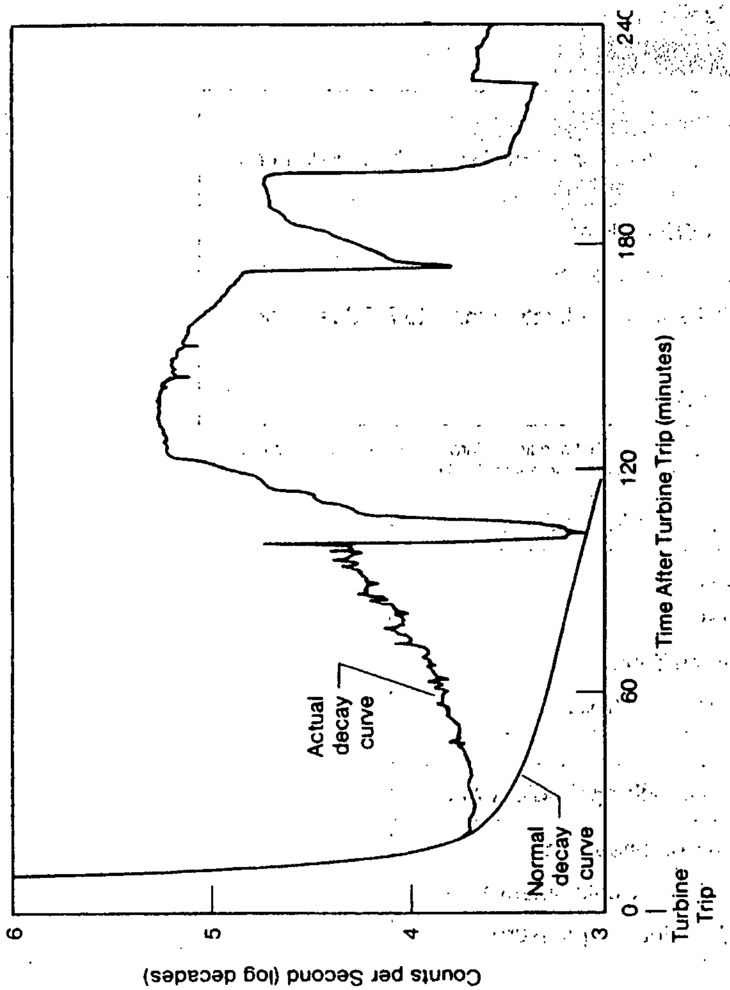
The containment radiation monitors high alarm is set at 1.0E+02 R/hr (ref 1). The 1.0E+02 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions while in the Refuel mode.

- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors such as Source Range Nuclear Instrumentation N-31 and N-32 can be used as a tool for making such determinations (ref 2). Figure C-1 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor.
- If water level indication is unavailable, Reactor Vessel inventory loss must be detected by Containment Sumps, Auxiliary Building Sump or RCDT level changes (ref. 3, 4, 5, 6, 7). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. P-6 Precautions, Limitations and Setpoints Nuclear Instrumentation System
3. RF-601 Fuel Handling Accident Instructions
4. O-2.3 Draining the Reactor Coolant System to < 84" but > 64"
5. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
6. UFSAR 5.1.3.6 Monitoring Reactor Coolant Leakage
7. UFSAR 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
8. NEI 99-01 CS1

**Figure C-1: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident**



**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged

**EAL:**

**CG3.1 General Emergency**

RCS level **cannot** be monitored with core uncovery indicated by **ANY** of the following for ≥ 30 min. (Note 4):

- Containment radiation R-29 or R-30 > 1.0E+02 R/hr
- Erratic Source Range Nuclear Instrumentation indication
- Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage

**AND**

**Any** Containment Challenge Indication, Table C-3

Note 4: The ED 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

| Table C-2 RCS Leakage Indications |
|---|
| • Containment Sump A |
| • Containment Sump B |
| • Auxiliary Building Sump Tank |
| • Reactor Coolant Drain Tank (RCDT) |

| Table C-3 Containment Challenge Indications |
|---|
| • Containment closure **not** established |
| • Hydrogen concentration in Containment ≥ 4% |
| • Unplanned rise in Containment pressure |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This ~~IC~~ EAL represents the inability to restore and maintain ~~RPV~~ RCS level to above the top of active fuel with containment challenged. Fuel damage is probable if ~~RPV~~ RCS level cannot be restored, as available decay heat will cause boiling, further reducing the ~~RPV~~ RCS level. With the Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers.

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

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include:

~~[BWRs] initial vessel level, shutdown heat removal system design~~

~~[PWRs]~~ mid-loop, reduced level/flange level, head in place, cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining

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

If Containment Closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to General Emergency would not occur.

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

~~[In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in Containment. However, Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.]~~

~~[For BWRs, the use of secondary containment radiation monitors should provide indication of increased release that may be indicative of a challenge to secondary containment. The site specific radiation monitor values should be based on the EOP "maximum safe values" because these values are easily recognizable and have an emergency basis.]~~

~~EAL #2~~

Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

~~[In the cold shutdown mode, normal RCS level and RPV level instrumentation systems will usually be available. In the refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.]~~

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

~~[This EAL should conservatively estimate a site specific dose rate setpoint indicative of core uncovery (ie., level at TOAF). For BWRs that do not have installed radiation monitors capable of indicating core uncovery, alternate site specific level indications of core uncovery should be used.]~~

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

Plant-Specific

In Refuel or Cold Shutdown mode, normal RCS level indication may be unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- As water level in the Reactor Vessel lowers, the dose rate above the core will rise. Containment radiation is indicated on R-29 and R-30. The dose rate due to this core shine should result in on-scale Containment radiation monitor indication and possible alarm. Assuming total draindown of the upper cavity, line-of-sight dose rates from a fully exposed upper internal package would be approximately 300 R/hr. The containment radiation monitors high alarm is set at 1.0E+02 R/hr (ref 1). The 1.0E+02 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions while in the Refuel mode.
- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors such as Source Range Nuclear Instrumentation N-31 and N-32 can be used as a tool for making such determinations (ref 2). Figure C-1 shows the response of the source

range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor.

- If water level indication is unavailable, Reactor Vessel inventory loss must be detected by Containment Sumps, Auxiliary Building Sump or RCDT level changes (ref. 3, 4, 5, 6, 7). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Three indications are associated with Containment challenges:

- Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in O-2.3.1A "Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation", provide a functional barrier to fission product release (ref 8). Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires that, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS bulk boiling.
- 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 Containment. However, Containment monitoring (CH-EPIP-CVH2) and/or sampling should be performed to verify this assumption. A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4.1% (rounded to 4%) by volume (ref. 9).

- Unplanned Containment pressure increases are not expected during Cold Shutdown or Refuel mode. The threshold is indicative of conditions challenging containment closure.

Definitions:

**Containment Closure**

The site-specific procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to Ginna, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in O-2.3.1A, provide a functional barrier to fission product release.

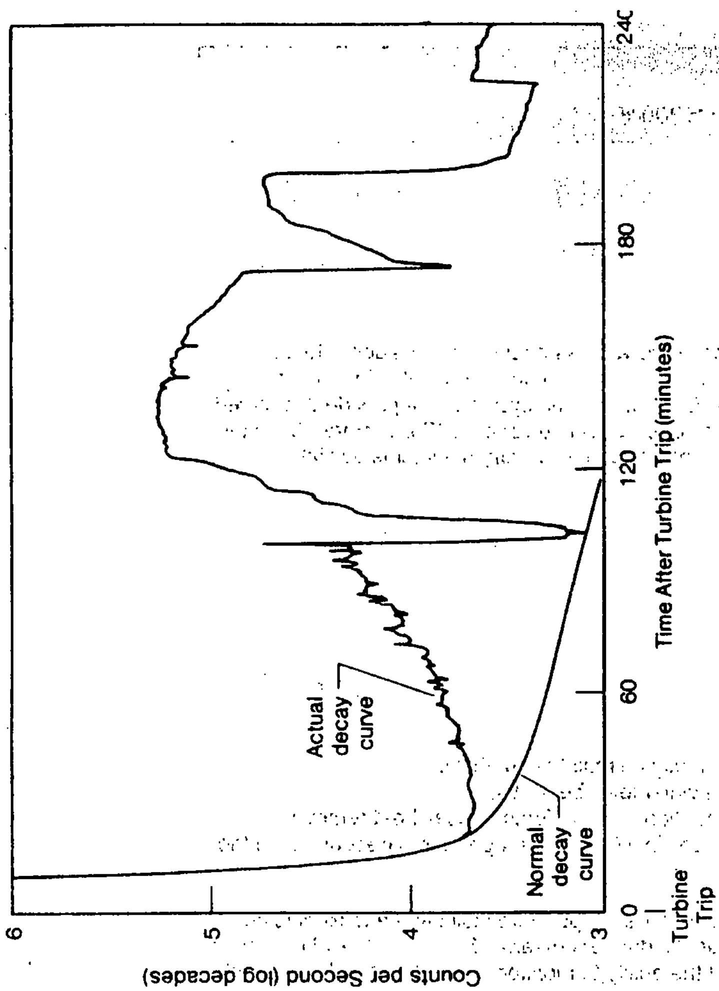
**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. P-6 Precautions, Limitations and Setpoints Nuclear Instrumentation System
3. RF-601 Fuel Handling Accident Instructions
4. O-2.3 Draining the Reactor Coolant System to < 84" but > 64"
5. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
6. UFSAR 5.1.3.6 Monitoring Reactor Coolant Leakage
7. UFSAR 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
8. O-2.3.1A Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation
9. SACRG-1 Severe Accident Control Room Guideline Initial Response
10. NEI 99-01 CG1

**Figure C-1: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident**



**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 4 – RCS Temperature

**Initiating Condition:** Unplanned loss of decay heat removal capability

**EAL:**

| | |
|---|---|
| **CU4.1** | **Unusual Event** |
| Unplanned event results in RCS temperature > 200°F | |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This ~~IC~~ EAL is ~~be~~ a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered.

[~~Entry into cold shutdown conditions may be attained within hours of operating at power. Entry into the refueling mode procedurally may not occur for typically 100 hours (site specific) or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the RPV (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). In addition, the operators should be able to monitor RCS temperature and RPV level so that escalation to the alert level via CA4 or CA1 will occur if required.~~]

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

[~~Unlike the cold shutdown mode,~~] ~~n~~Normal means of core temperature indication and RCS level indication may not be available in the Refuel mode. Redundant means of ~~RPV~~ RCS level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. ~~However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, EAL 2 would result in declaration of a NOUE if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication.~~ Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1~~4~~ based on exceeding its temperature duration or pressure criteria.

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F) (ref. 1). These include (ref. 2):

- The average of T0409A ($T_{HOT}$) and T0410B ($T_{COLD}$) for forced circulation with A RCP pump running
- The average T0410B ($T_{COLD}$) **AND** either T0410A **OR** incore thermocouples for $T_{HOT}$ for forced circulation with B RCP pump running
- T0630 Residual Heat Removal Pump Discharge Header
- Incore Temperatures

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. Technical Specifications Table 1.1-1
2. O-2.2 Plant Shutdown from Hot Shutdown to Cold Conditions
3. NEI 99-01 CU4

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 4 – RCS Temperature

**Initiating Condition:** Unplanned loss of decay heat removal capability

**EAL:**

| | |
|---|---|
| **CU4.2** | **Unusual Event** |
| Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 4) | |

Note 4: The ED 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

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This ~~IC~~ EAL is ~~be~~ a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered.

~~[*Entry into cold shutdown conditions may be attained within hours of operating at power. Entry into the refueling mode procedurally may not occur for typically 100 hours (site specific) or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the RPV (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). In addition, the operators should be able to monitor RCS temperature and RPV level so that escalation to the alert level via CA4 or CA1 will occur if required.*]~~

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

~~[*Unlike the cold shutdown mode,*]~~ ~~n~~Normal means of core temperature indication and RCS level indication may not be available in the Refuel mode. Redundant means of ~~RPV~~ RCS level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown or Refuel modes, ~~EAL 2~~this EAL would result in declaration of a ~~NO~~UE if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication.

Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.14 based on exceeding its temperature criteria.

Plant-Specific

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refuel mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refuel mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. Reactor Vessel water level is normally monitored in Refuel mode using the following instruments (ref. 1,5,6):

- Loop A Level Indicator LIT-432A
- Loop B Level Indicator LIT-432B
- Loop B Sightglass
- Cavity Water Level

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F) (ref. 2). These include (ref. 3):

- The average of T0409A ($T_{HOT}$) and T0410B ($T_{COLD}$) for forced circulation with A RCP pump running
- The average of T0410B ($T_{COLD}$) **AND** either T0410A **OR** incore thermocouples for $T_{HOT}$ for forced circulation with B RCP pump running
- T0630 Residual Heat Removal Pump Discharge Header
- Incore Temperatures

**Ginna Basis Reference(s):**

1. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
2. Technical Specifications Table 1.1-1
3. O-2.2 Plant Shutdown from Hot Shutdown to Cold Conditions
4. NEI 99-01 CU4

5. O-15.1 Administrative Requirement Checklist for Entry to Mode 6 and Refueling Conditions
6. O-6.13 Daily Surveillance Log

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 4 – RCS Temperature

**Initiating Condition:** Inability to maintain plant in cold shutdown

**EAL:**

| **CA4.1** | **Alert** |
|---|---|
| An unplanned event results in **EITHER**: | |
| RCS temperature > 200°F for > Table C-4 duration | |
| **OR** | |
| RCS pressure increase > 10 psi due to an unplanned loss of decay heat removal capability (this condition is **not** applicable in solid plant conditions) | |

**Table C-4 RCS Reheat Duration Thresholds**

| RCS Status | Containment Closure Status | Duration |
|---|---|---|
| Intact **AND not** reduced inventory | N/A | 60 min.* |
| **Not** intact **OR** reduced inventory | Established | 20 min.* |
| | **Not established** | 0 min. |

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

**Mode Applicability:**

5 - Cold Shutdown, 6 – Refuel

**Basis:**

Generic

~~For EAL 1, t~~The RCS Reheat Duration Thresholds table addresses complete loss of functions required for core cooling for greater than 60 minutes during refueling and cold shutdown modes when RCS integrity is established. ~~[RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this condition is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment.]~~The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

The RCS Reheat Duration Thresholds table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during refueling and cold shutdown modes when

Containment Closure is established but RCS integrity is not established or RCS inventory is reduced ~~[(e.g., mid loop operation in PWRs)]. [As discussed above, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).]~~. The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. ~~[The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed later in this basis) and is believed to be conservative given that a low pressure Containment barrier to fission product release is established.]~~

Finally, complete loss of functions required for core cooling during refuel~~ing~~ and cold shutdown modes when neither Containment Closure nor RCS integrity are established is addressed. ~~[RCS integrity is in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).~~ No delay time is allowed because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.]

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

~~In EAL 2, t~~The 10 psig pressure increase addresses situations where~~,~~ due to high decay heat loads, the time provided to restore temperature control~~,~~ should be less than 60 minutes. The RCS pressure setpoint was chosen ~~should be 10 psi o~~because it is~~r~~ the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psig.

Escalation to Site Area Emergency would be via EAL CS3.1 should boiling result in significant ~~RPV~~ Reactor Vessel level loss leading to core uncovery.

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

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

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F) (ref. 1). These include (ref. 2):

- The average of T0409A ($T_{HOT}$) and T0410B ($T_{COLD}$) for forced circulation with A RCP pump running
- The average of T0410B ($T_{COLD}$) **AND** either T0410A **OR** incore thermocouples for $T_{HOT}$ for forced circulation with B RCP pump running
- T0630 Residual Heat Removal Pump Discharge Header

Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in O-2.3.1A "Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation" (ref. 3), provide a functional barrier to fission product release. Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS boiling.

Reduced Inventory (administrative) is defined as RCS level less than 64 in. on the RCS Loop indicators (ref. 4).

The pressure rise of greater than 10 psig implies an RCS temperature in excess of the Technical Specification cold shutdown limit (200°F) for which this EAL would otherwise permit up to sixty minutes to restore RCS cooling before declaration of an Alert (RCS intact). This EAL therefore covers situations in which it is determined that, due to high decay heat loads, the time provided to reestablish temperature control should be less than sixty minutes (as indicated by significant RCS re-pressurization).

Pressure indicator PI-420 Rx Clnt Loop Lo Rng Press is capable of measuring pressure changes of 10 psig ~~(ref. 5)~~. This represents the visual resolution of the device, with the smallest scale increment of 10 psig (Basis: Walkdown). Escalation to a Site Area

Emergency would be under EAL CS2.1 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

Definitions:

**Containment Closure**

The site-specific procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to Ginna, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in O-2.3.1A, provide a functional barrier to fission product release.

**RCS Intact**

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

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. Technical Specifications Table 1.1-1
2. O-2.2 Plant Shutdown from Hot Shutdown to Cold Conditions
3. O-2.3.1A Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation
4. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
5. ~~CPI-PRESS-420 Calibration of Reactor Coolant System Pressure Channel 420 Rack Instrumentation~~

~~6.~~5. NEI 99-01 CA4

**Category:** C – Cold Shutdown / Refueling System Malfunction
**Subcategory:** 5 – Communications
**Initiating Condition:** Loss of **all** onsite or offsite communications capabilities

**EAL:**

**CU5.1 Unusual Event**

Loss of **all** Table C-5 onsite (internal) communication methods affecting the ability to perform routine operations

**OR**

Loss of **all** Table C-5 offsite (external) communication methods affecting the ability to perform offsite notifications

**Table C-5 Communications Systems**

| System | Onsite (internal) | Offsite (external) |
|---|---|---|
| Commercial phone system | X | X |
| Direct Dial POTS Lines (Blue Phones) | X | X |
| Plant Page Party system | X | |
| Radios/Walkie Talkies | X | |
| FTS 2001 telephone system (ENS, HPN) | | X |
| Control Room Hard Wired Satellite Phone | | X |
| Control Room Emergency Cell Phone | | X |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel, D - Defueled

**Basis:**

Generic

The purpose of this ~~IC~~ EAL~~and its associated EALs~~ is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant issues. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to off-site locations, etc.) are being utilized to make communications possible.

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

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

Plant-Specific

Onsite/offsite communications systems are listed in Table C-2 (ref. 1, 2).

This EAL is the cold condition equivalent of the hot condition EAL SU6.1.

**Ginna Basis Reference(s):**

1. A-56 Communication Systems at Ginna Station
2. ER-COMM.1 Loss of Communications
3. NEI 99-01 CU6

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 6 – Inadvertent Criticality

**Initiating Condition:** Inadvertent criticality

**EAL:**

| | |
|---|---|
| **CU6.1** | **Unusual Event** |
| An unplanned sustained positive startup rate observed on nuclear instrumentation | |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This ~~IC~~ EAL addresses criticality events that occur in Cold Shutdown or Refuel modes ~~[(NUREG 1449, Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States)]~~ such as fuel mis-loading events and inadvertent dilution events. This ~~IC~~ EAL indicates a potential degradation of the level of safety of the plant, warranting a N~~O~~UE classification.

~~[This condition can be identified using period monitors/startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive periods/startup rates from planned fuel bundle or control rod movements during core alteration for PWRs and BWRs. These short term positive periods/startup rates are the result of the increase in neutron population due to subcritical multiplication.]~~

Escalation would be by Emergency Director j~~J~~udgment.

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the rise in neutron population due to subcritical multiplication. Short-term positive startup rates can also be due to welding activities.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. NEI 99-01 CU8

**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.

The events of this category pertain to the following subcategories:

1. Natural or Destructive Phenomena

Natural events include hurricanes, earthquakes or tornadoes that have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety. Non-naturally occurring events that can cause damage to plant facilities include aircraft crashes, missile impacts, etc.

2. Fire or Explosion

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

3. Hazardous Gas

Non-naturally occurring events that can cause damage to plant facilities and include toxic, asphyxiant, corrosive or flammable gas leaks.

4. Security

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

5. 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.

6. 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 Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

**HU1.1 Unusual Event**

Seismic event identified by **ANY two** of the following:

- Red LED event indicator on Kinemetrics ETNA Digital Recorder indicates seismic event detected
- Earthquake felt onsite
- National Earthquake Information Center (Note 6)

Note 6: 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 Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude**.

**Mode Applicability:**

All

**Basis:**

Generic

The~~se EALs are~~is EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

~~EAL #1~~

Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.

As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.

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

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

~~EAL #2~~

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

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

~~Escalation of this emergency classification level, if appropriate, would be based on VISIBLE DAMAGE, or by other in plant conditions, via HA1.~~

~~EAL #3~~

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

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

~~Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via HA1, or by other plant conditions.~~

~~EAL #4~~

~~This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.~~

~~Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via HU2 and HU3.~~

~~This EAL is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non safety related equipment.~~

~~Escalation of this emergency classification level, if appropriate, would be to HA1 based on damage done by PROJECTILES generated by the failure or by the radiological releases for a BWR, or in conjunction with a steam generator tube rupture, for a PWR. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs.~~

~~EAL #5~~

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

~~[Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.).]~~

Plant-Specific

A strong motion accelerograph is installed in the subbasement of the intermediate building at elevation 237 ft (ref. 1).

Ginna seismic instrumentation actuates upon sensing any ground motion greater than 0.01g. Registration of a tremor > 0.01g is indicated by a red light on the event indicator at the bottom of the accelerograph case (ref. 2, 3, 4).

The National Earthquake Information Center (NEIC) can confirm seismic activity in the vicinity of Ginna. 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 R. E. Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude** (ref. 5).

**Ginna Basis Reference(s):**

1. UFSAR Section 3.7.4 Seismic Instrumentation
2. ER-SC.4 Earthquake Emergency Plan
3. CPI-ACCELEROGRAPH-51 Functional Check of Kinemetrics Strong Motion Accelerograph
4. VTD-K3356-4104 Kinematics, ETNA Strong Motion Accelerograph Schematics
5. USAR Section 2.1.1 Site Location and Description
6. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

| | |
|---|---|
| **HU1.2** | **Unusual Event** |
| Tornado striking within Protected Area boundary | |
| **OR** | |
| Sustained high winds > 75 mph | |

**Mode Applicability:**

All

**Basis:**

Generic

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

~~EAL #1~~

~~Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.~~

~~As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.~~

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

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

~~EAL #2~~

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

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

Escalation of this emergency classification level, if appropriate, would be based on visible damage, or by other in plant conditions, via EAL HA1.2.

~~EAL #3~~

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

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

~~Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via HA1, or by other plant conditions.~~

~~EAL #4~~

~~This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.~~

~~Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build-up are appropriately classified via HU2 and HU3.~~

~~This EAL is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.~~

~~Escalation of this emergency classification level, if appropriate, would be to HA1 based on damage done by PROJECTILES generated by the failure or by the radiological releases for a BWR, or in conjunction with a steam generator tube rupture, for a PWR. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs.~~

~~EAL #5~~

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

~~[*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call outs, etc.).*]~~

Plant-Specific

All Class 1 structures are designed for a wind velocity of 75 mph assuming FSAR "severe environmental loading" conditions (ref. 1).

Wind speed can be measured up to 100 mph on the 250' and 150' wind speed recorder 'A' (ref. 2). Sustained winds are the five-minute average wind speed.

The Protected Area Boundary is depicted in Drawing 33013-2722 (ref. 3)

Definitions:

**Protected Area**

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

**Ginna Basis Reference(s):**

1. UFSAR Section 3.3.2.1.4 Wind and Tornado Loadings - Input Load Criteria
2. CPI-MET-250 Calibration of Ginna Station Meteorological Wind Speed and Wind Direction Translator Cards
3. Drawing 33013-2722 Residential AC Power Distribution Circuit - Site Layout
4. ER-SC.1 Adverse Weather Plan
5. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

**HU1.3 Unusual Event**

Internal flooding that has the potential to affect **ANY** safety-related structure, system, or component required by Technical Specifications for the current operating mode in **ANY** Table H-1 area

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

~~These~~ This EAL~~s are~~ is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

~~EAL #1~~

~~Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.~~

~~As defined in the EPRI sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.~~

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

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

~~EAL #2~~

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

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

~~Escalation of this emergency classification level, if appropriate, would be based on VISIBLE DAMAGE, or by other in plant conditions, via HA1.~~

~~EAL #3~~

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

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

Escalation of this emergency classification level, if appropriate, would be based visible damage via EAL HA1.3, or by other plant conditions.

~~EAL #4~~

~~This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.~~

~~Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via HU2 and HU3.~~

~~This EAL is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.~~

~~Escalation of this emergency classification level, if appropriate, would be to HA1 based on damage done by PROJECTILES generated by the failure or by the radiological releases for a BWR, or in conjunction with a steam generator tube rupture, for a PWR. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs.~~

~~EAL #5~~

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

~~[*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call outs, etc.).*]~~

Plant-Specific

This threshold addresses the effect of flooding caused by internal events such as component failures, Circulating, Component Cooling or Service Water line ruptures, equipment misalignment, fire suppression system actuation, and outage activity mishaps.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of its removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

This threshold addresses events that may have resulted in a Safe Shutdown Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Safe Shutdown Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern (ref. 1, 2).

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

| **HU1.4** | **Unusual Event** |
|---|---|
| Turbine failure resulting in casing penetration or damage to turbine or generator seals | |

**Mode Applicability:**

All

**Basis:**

Generic

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

~~EAL #1~~

~~Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.~~

~~As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.~~

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

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

~~EAL #2~~

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

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

~~Escalation of this emergency classification level, if appropriate, would be based on VISIBLE DAMAGE, or by other in plant conditions, via HA1.~~

~~EAL #3~~

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

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

~~Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via HA1, or by other plant conditions.~~

~~EAL #4~~

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.

Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via EAL HU2.1 and EAL HU3.1.

This EAL is consistent with the definition of a ~~NO~~UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of this emergency classification level, if appropriate, would be to EAL HA1.4 based on damage done by PROJECTILES generated by the failure or ~~by the radiological releases for a BWR, or~~ in conjunction with a steam generator tube rupture~~, for a PWR~~. These latter events would be classified by the Category R~~radiological ICs~~ EALs or ~~Fission Product Barrier~~Category F ~~ICs~~EALs.

~~EAL #5~~

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

~~[*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.)*.]~~

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external projectiles will be released. These ejected projectiles may impact various plant structures, including those housing safety related equipment.

Failure of turbine or generator seals may be indicated by a loss of seal oil pressure or loss of condenser vacuum (ref. 1).

**Ginna Basis Reference(s):**

1. ER-SC.8 Turbine Blade Failure and Missile Emergency Plan
2. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety
**Subcategory:** 1 – Natural or Destructive Phenomena
**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area
**EAL:**

| | |
|---|---|
| **HU1.5** | **Unusual Event** |
| Deer Creek flooding over entrance road bridge hand rail<br>**OR**<br>Lake level > 252 ft<br>**OR**<br>Screen House Suction Bay water level < 17 ft or < 15.5 ft by manual level measurement | |

**Mode Applicability:**

All

**Basis:**

Generic

~~These~~ This EAL~~s are~~ is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

~~EAL #1~~

~~Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.~~

~~As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.~~

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

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

~~EAL #2~~

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

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

~~Escalation of this emergency classification level, if appropriate, would be based on VISIBLE DAMAGE, or by other in plant conditions, via HA1.~~

~~EAL #3~~

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

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

~~Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via HA1, or by other plant conditions.~~

~~EAL #4~~

~~This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.~~

~~Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via HU2 and HU3.~~

~~This EAL is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.~~

~~Escalation of this emergency classification level, if appropriate, would be to HA1 based on damage done by PROJECTILES generated by the failure or by the radiological releases for a BWR, or in conjunction with a steam generator tube rupture, for a PWR. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs.~~

~~EAL #5~~

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

~~[*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call outs, etc.).*]~~

**Plant-Specific**

This threshold addresses high and low lake water level conditions that could be a precursor of more serious events.

Ginna plant grade is generally at 270 ft mean sea level (msl) except the area between the lake and Turbine Building which is at grade 253 ft msl. Lake water level > 253.28 ft msl corresponds to plant design levels (ref. 1). A lake level of 252 ft has been selected for this threshold to be anticipatory of exceeding design flood levels and is the level at which flood control actions are procedurally taken (ref. 2).

Flooding in Deer Creek above the plant entrance handrails will ultimately result in water accumulation in the Turbine Building and Screenhouse (ref. 2). This may preclude emergency response personnel access and egress.

High lake level may be determined using markers attached to a metal pole mounted on the discharge canal bridge upstream of the submarine net. The high level markers are at lake levels of 252 ft and 253 ft (ref. 2).

The Screenhouse Lo-Lo level alarm actuates at 19' indicated (ref. 3). When Screenhouse Suction Bay water level drops to 17.0 ft indicated (this corresponds to a level of 15.5' measured manually) increased Control Room monitoring is initiated. This level has been selected for this threshold to be anticipatory of a potential loss of service water system pump suction at 16.0 ft (ref. 4).

**Ginna Basis Reference(s):**

1. UFSAR Section 3.4.1 Flood Protection
2. ER-SC.2 High Water (Flood) Plan
3. AR-I-9 Screen House Lo-Lo Level 19'
4. ER-SC.3 Low Screenhouse Water Level
5. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.1 Alert**

**EITHER**:

Confirmation of an earthquake of an intensity > 0.08 g per ER-SC.4 Earthquake Emergency Plan

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component

**AND**

Earthquake confirmed by **EITHER**:

Earthquake felt in onsite

**OR**

National Earthquake Information Center (Note 6)

Note 6: 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 Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude.**

**Mode Applicability:**

All

**Basis:**

Generic

These EALs escalate from HU1.1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction ~~ICs~~EALs.

~~EALs #2 - #5~~

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

~~EAL #1~~

Seismic events of this magnitude can result in a vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

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

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

~~EAL #2~~

~~This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant.~~

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

~~EAL #3~~

~~This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.~~

~~Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.~~

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

~~EAL #4~~

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

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

~~EAL #5~~

~~This EAL addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.~~

~~EAL #6~~

~~This EAL addresses other site specific phenomena that result in VISIBLE DAMAGE to VITAL AREAS or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant (such as hurricane, flood, or seiche) that can also be precursors of more serious events.~~

~~[*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call outs, etc.).*]~~

Plant-Specific

This EAL is based on the UFSAR design basis operating earthquake of 0.08 g acceleration (ref. 1). Seismic events of this magnitude can cause damage to plant safety functions.

Ginna seismic instrumentation actuates upon sensing any seismic activity (ref. 2, 3, 4).

The method of detection of an earthquake greater than OBE intensity relies on either:

- analysis of the Ginna strong motion accelerograph located in the Intermediate Building by plant I&C and plant Engineering (ref. 3)

  **OR**

- by actual indications of degraded safe shutdown system performance

confirmed by either shift operators on duty in the Control Room determining that ground motion was felt, or corroborated by the NEIC.

The National Earthquake Information Center (NEIC) can confirm seismic activity in the vicinity of Ginna. 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 R. E.

Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude** (ref. 5).

Definitions:

**Safety-Related Structures, Systems and Components** (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.

**Ginna Basis Reference(s):**

1. UFSAR Section 3.7.1.3 Design Response Spectra
2. ER-SC.4 Earthquake Emergency Plan
3. CPI-ACCELEROGRAPH-51 Functional Check of Kinemetrics Strong Motion Accelerograph
4. VTD-K3356-4104 ETNA Strong Motion Accelerograph Schematics
5. USAR Section 2.1.1 Site Location and Description
6. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.2 Alert**

Tornado striking or sustained high winds > 75 mph resulting in **EITHER**:

Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

~~These~~ This EAL~~s~~ escalates from HU1.2 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction ~~ICs~~EALs.

~~EALs #2 - #5~~

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

~~EAL #1~~

~~Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.~~

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

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

~~EAL #2~~

This EAL is based on a tornado striking (touching down) or high winds that have caused visible damage to structures containing functions or systems required for safe shutdown of the plant.

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

~~EAL #3~~

~~This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.~~

~~Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.~~

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

~~EAL #4~~

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

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

~~EAL #5~~

~~This EAL addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.~~

~~EAL #6~~

~~This EAL addresses other site specific phenomena that result in VISIBLE DAMAGE to VITAL AREAS or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant (such as hurricane, flood, or seiche) that can also be precursors of more serious events.~~

[~~*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call outs, etc.).*~~]

Plant-Specific

All Class 1 structures are designed for a wind velocity of 75 mph assuming FSAR "severe environmental loading" conditions (ref. 1). This EAL is based on the structural design basis of 75 mph or impact by tornado. Wind loads of this magnitude can cause damage to safety functions.

Wind speed can be measured up to 100 mph on the 250' and 150' wind speed recorder 'A' (ref. 2). Sustained winds are the five-minute average wind speed.

The Protected Area Boundary is depicted in Drawing 33013-2722 (ref. 3).

This threshold addresses events that may have resulted in a Safe Shutdown Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Safe Shutdown Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern. The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Safe Shutdown Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold.

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 4, 5).

Definitions:

**Visible Damage**

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

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. UFSAR Section 3.3.2.1.4 Wind and Tornado Loadings - Input Load Criteria
2. CPI-MET-250 Calibration of Ginna Station Meteorological Wind Speed and Wind Direction Translator Cards
3. Drawing 33013-2722 Residential AC Power Distribution Circuit - Site Layout
4. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
5. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
6. ER-SC.1 Adverse Weather Plan
7. NEI 99-01 HA1

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**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.3 Alert**

Internal flooding in **ANY** Table H-1 area resulting in **EITHER**:

An electrical shock hazard that precludes access to operate or monitor **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

~~These EALs escalate from HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here~~

~~is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.~~

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EAL~~IC~~s.

~~EALs #2 - #5~~

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

~~EAL #1~~

~~Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.~~

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

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

~~EAL #2~~

~~This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant.~~

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

~~EAL #3~~

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

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

~~EAL #4~~

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

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

~~EAL #5~~

~~This EAL addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.~~

~~EAL #6~~

~~This EAL addresses other site specific phenomena that result in VISIBLE DAMAGE to VITAL AREAS or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant (such as hurricane, flood, or seiche) that can also be precursors of more serious events.~~

[~~*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.).*~~]

Plant-Specific

This threshold addresses the effect of flooding caused by internal events such as component failures such as Circulating, Component Cooling or Service Water line ruptures, equipment misalignment, fire suppression system actuation, steam leaks or outage activity mishaps.

Safe Shutdown Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern (ref. 1, 2).

Uncontrolled internal flooding that has degraded safety shutdown equipment or created a safety hazard precluding access necessary for the safe operation or monitoring of safety equipment warrants declaration of an Alert.

Definitions:

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.4 Alert**

Turbine failure-generated projectiles resulting in **EITHER**:

Visible damage to or penetration of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

Th~~ese~~is EALs escalates from HU1.4 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction ~~ICs~~EALs.

~~EALs #2 - #5~~

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

~~EAL #1~~

~~Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.~~

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

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

~~EAL #2~~

~~This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant.~~

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

~~EAL #3~~

~~This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.~~

~~Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.~~

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

~~EAL #4~~

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

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

~~EAL #5~~

~~This EAL addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.~~

~~EAL #6~~

~~This EAL addresses other site specific phenomena that result in VISIBLE DAMAGE to VITAL AREAS or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant (such as hurricane, flood, or seiche) that can also be precursors of more serious events.~~

~~[*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.).*]~~

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external projectiles will be released. These ejected projectiles may impact various plant structures, including those housing safety related equipment.

Failure of turbine or generator seals may be indicated by a loss of seal oil pressure or loss of condenser vacuum (ref. 1).

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 2, 3).

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

~~EAL #2~~

~~This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant.~~

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

~~EAL #3~~

~~This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.~~

~~Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.~~

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

~~EAL #4~~

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

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

~~EAL #5~~

~~This EAL addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.~~

~~EAL #6~~

This EAL addresses other site specific phenomena that result in visible damage to vital areas or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant ~~(such as hurricane, flood, or seiche)~~ that can also be precursors of more serious events.

~~[Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.).]~~

Plant-Specific

This threshold covers high and low water level conditions that may have resulted in a plant safe shutdown area being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

Ginna plant grade is generally at 270 ft mean sea level (msl) except the area between the lake and Turbine Building which is at grade 253 ft msl. Lake water level > 253.28 ft msl corresponds to plant design levels (ref. 1). A lake level of 253 ft has been selected for this threshold to be indicative of exceeding design flood levels (ref. 2).

High lake level may be determined using markers attached to a metal pole mounted on the discharge canal bridge upstream of the submarine net. The high level markers are at lake levels of 252 ft and 253 ft (ref. 2).

The Screenhouse Lo-Lo level alarm actuates at 19' indicated (ref. 3). If indicated service water pump bay level drops below 16 ft (this corresponds to a lake level of 14.5' measured manually) the service water pumps are declared inoperable. This level has been selected for this threshold to be indicative of a loss of service water system pump suction (ref. 4).

**Ginna Basis Reference(s):**

1. UFSAR Section 3.4.1 Flood Protection
2. ER-SC.2 High Water (Flood) Plan
3. AR-I-9 Screen House Lo-Lo Level 19'
4. ER-SC.3 Low Screenhouse Water Level
5. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.6 Alert**

Vehicle crash resulting in **EITHER**:

Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building |
| • Auxiliary Building |
| • Control Building |
| • Intermediate Building |
| • Emergency Diesel Building(s) |
| • SAFW Building |
| • Screenhouse |
| • Cable Tunnel |
| • Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

~~These EALs escalate from HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance.~~ The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction ~~ICs~~EALs.

~~EALs #2 - #5~~

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

~~EAL #1~~

~~Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.~~

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

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

~~EAL #2~~

~~This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant.~~

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

~~EAL #3~~

~~This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.~~

~~Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.~~

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

Definitions:

**Projectile**

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

**Visible Damage**

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

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. ER.SC-8 Turbine Blade Failure and Missile Emergency Plan
2. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
3. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
4. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

| | |
|---|---|
| **HA1.5** | **Alert** |
| Lake level > 253 ft<br>**OR**<br>Screen House Suction Bay water level ≤ 16 ft or < 14.5 ft by manual level measurement | |

**Mode Applicability:**

All

**Basis:**

Generic

~~These EALs escalate from HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.~~

~~Escalation of this emergency classification level, if appropriate, would be based on System Malfunction ICs.~~

~~EALs #2 - #5~~

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

~~EAL #1~~

~~Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.~~

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

~~EAL #4~~

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

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

~~EAL #5~~

This EAL addresses vehicle crashes within the Protected Area that results in visible damage to vital areas or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.

~~EAL #6~~

~~This EAL addresses other site specific phenomena that result in VISIBLE DAMAGE to VITAL AREAS or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant (such as hurricane, flood, or seiche) that can also be precursors of more serious events.~~

~~[*Sites subject to severe weather as defined in the NUMARC station blackout initiatives should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call outs, etc.).*]~~

Plant-Specific

This EAL is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. Vehicle types include automobiles, aircraft, trucks, cranes, forklifts, waterborne craft, etc.

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 1, 2).

Definitions:

**Visible Damage**

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

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire within the Protected Area **not** extinguished within 15 min. of detection or explosion within the Protected Area

**EAL:**

| **HU2.1** | **Unusual Event** |
|---|---|
| Fire **not** extinguished within 15 min. of Control Room notification or verification of a Control Room fire alarm in **ANY** Table H-1 area or Turbine Building (Note 4) | |

Note 4: The ED 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.

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses the magnitude and extent of fires ~~or EXPLOSIONS~~ that may be potentially significant precursors of damage to safety systems. It addresses the FIRE ~~/ EXPLOSION~~, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and report by plant personnel or sensor alarm indication.

~~EAL #1~~

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems.

As used here, notification is visual observation and report by plant personnel or sensor alarm indication.

The 15-minute period to extinguish the fire begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm.

Determination of a valid fire detection system alarm includes actions that can be taken within the Control Room or at nearby Fire Panels to determine that the alarm is not spurious. These actions include the use of direct or indirect indications such as redundant alarms or instrumentation readings associated with the area to ensure the alarm is not spurious and is an indication of a fire. An alarm verified in this manner is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm. If the alarm cannot be verified by redundant Control Room or nearby Fire Panel indications, notification from the field that a fire exists would be required to start the 15-minute classification and fire extinguishment clocks. ~~The 15 minute time period begins with a credible notification that a fire is occurring, or indication of a fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken within the control room or other nearby site specific location to ensure that it is not spurious. An alarm is assumed to be an indication of a fire unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.~~

The intent of this 15 minute duration is to size the fire and to discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket).

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

~~EAL #2~~

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

~~No attempt is made to assess the actual magnitude of the damage. The occurrence of the EXPLOSION is sufficient for declaration.~~

~~The Emergency director also needs to consider any security aspects of the EXPLOSION, if applicable.~~

~~Escalation of this emergency classification level, if appropriate, would be based on HA2.~~

## Plant-Specific

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 1, 2). The Turbine Building is included because it is immediately adjacent to one or more Table H-1 areas and a fire within the Turbine Building may potentially impact safe shutdown equipment should the fire not be controlled.

Definitions:

**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.

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume 1 Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HU2

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire within the Protected Area **not** extinguished within 15 min. of detection or explosion within the Protected Area

**EAL:**

| | |
|---|---|
| **HU2.2** | **Unusual Event** |
| Explosion within the Protected Area | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses the magnitude and extent of ~~FIRES or~~ explosions that may be potentially significant precursors of damage to safety systems. It addresses the ~~FIRE /~~ explosion, and not the degradation in performance of affected systems that may result.

~~As used here, detection is visual observation and report by plant personnel or sensor alarm indication.~~

~~EAL #1~~

~~The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken within the control room or other nearby site specific location to ensure that it is not spurious. An alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.~~

~~The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket).~~

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

~~EAL #2~~

This EAL addresses only those explosions of sufficient force to damage permanent structures or equipment within the Protected Area.

No attempt is made to assess the actual magnitude of the damage. The occurrence of the explosion is sufficient for declaration.

The Emergency Director also needs to consider any security aspects of the explosion, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA2.1.

Plant-Specific

While some explosions may also result in fires that exceed EAL HU2.1, no fire is necessary to declare an emergency in the event of an explosion. If a fire also occurs as a result or with an explosion, declare the Unusual Event based on the explosion and monitor the progress of the fire for potential escalation due to fire damage.

Definitions:

**Explosion**

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

**Protected Area**

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

**Ginna Basis Reference(s):**

1. Drawing 33013-2722 Residential AC Power Distribution Circuit - Site Layout
2. NEI 99-01 HU2

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown

**EAL:**

**HA2.1 Alert**

Fire or explosion resulting in **EITHER**:

Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building |
| • Auxiliary Building |
| • Control Building |
| • Intermediate Building |
| • Emergency Diesel Building(s) |
| • SAFW Building |
| • Screenhouse |
| • Cable Tunnel |
| • Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

Visible damage is used to identify the magnitude of the fire or explosion and to discriminate against minor fires and explosions.

The reference to structures containing safety systems or components is included to discriminate against fires or explosions in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the fire or explosion was large enough to cause damage to these systems.

The use of visible damage should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency ~~Director~~ with the resources needed to perform detailed damage assessments.

The Emergency Director also needs to consider any security aspects of the explosion.

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

Escalation of this emergency classification level, if appropriate, will be based on EALs in ~~System Malfunctions~~Category S, ~~Fission Product Barrier Degradation~~Category F or ~~Abnormal Rad Levels / Radiological Effluent ICs~~Category R.

Plant-Specific

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 1, 2).

Definitions:

**Explosion**

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

**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.

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HA2

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 3 – Hazardous Gas

**Initiating Condition:** Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

**EAL:**

| **HU3.1** | **Unusual Event** |
|---|---|
| Release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This ~~IC~~ EAL is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

Plant-Specific

Normal plant operations is defined to mean activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

Definitions:

**Normal Plant Operations**

Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

**Ginna Basis Reference(s):**

1. NEI 99-01 HU3

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 3 – Hazardous Gas

**Initiating Condition:** Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

**EAL:**

| **HU3.2** | **Unusual Event** |
|---|---|
| Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event | |

**Mode Applicability:**

All

**Basis:**

Generic

~~This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect NORMAL PLANT OPERATIONS.~~

~~The fact that SCBA may be worn does not eliminate the need to declare the event.~~

~~This IC is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.~~

~~An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.~~

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

Plant-Specific

None

**Ginna Basis Reference(s):**

1. NEI 99-01 HU3

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 3 – Hazardous Gas

**Initiating Condition:** Access to a Vital Area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor

**EAL:**

**HA3.1 Alert**

Access to **ANY** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of **ANY** safety-related structure, system, or component (Note 5)

Note 5: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should **not** be declared as it will have **no** adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

<u>Generic</u>

Gases in a Vital Area can affect the ability to safely operate or safely shutdown the reactor.

The fact that SCBA may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases.

This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If access is not required at the time the unsafe concentrations exist in the affected area or ~~i~~If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gasses which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on ~~System Malfunctions~~EALs in Category S, Category F or Category R~~, Fission Product Barrier Degradation or Abnormal Rad Levels / Radioactive Effluent ICs~~.

Plant-Specific

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 1, 2).

If hazardous gas concentration in a Table H-1 area restricts access but the equipment is not required to be operable or will not be required to operate before access can be reestablished (e.g., fans are ventilating the area), this EAL should not be declared.

Definitions:

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HA3

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

**Initiating Condition:** Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant

**EAL:**

| | |
|---|---|
| **HU4.1** | **Unusual Event** |

A security condition that does **not** involve a hostile action as reported by Security Shift Supervision

**OR**

A credible site-specific security threat notification

**OR**

A validated notification from NRC providing information of an aircraft threat

**Mode Applicability:**

All

**Basis:**

Generic

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as hostile actions are classifiable under EAL HA4.1, EAL HS4.1 and EAL ~~HG1~~HG4.1.

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

~~EAL #1~~First Condition

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

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

~~EAL #2~~Second Condition

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the ~~Notification of an~~ Unusual Event.

The determination of "credible" is made through use of information found in the Ginna Safeguards Contingency Plan ~~site specific Safeguards Contingency Plan~~.

~~EAL #3~~Third Condition

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

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level ~~would be~~ via EAL HA4.1 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

Plant-Specific

If the Security Shift Supervisor determines that a threat notification is credible, the Security Shift Supervisor will notify the Operations Shift Manager that a "Credible Threat" condition exists for Ginna. Generally, Ginna Security procedures address standard practices for determining credibility. The three main criteria for determining credibility are: technical feasibility, operational feasibility, and resolve. For Ginna, a validated notification delivered by the FBI, NRC or similar agency is treated as credible.

Definitions:

**Airliner/Large Aircraft**

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

**Hostile Action**

An act toward Ginna 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 Ginna. 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).

**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.

**Ginna Basis Reference(s):**

1. Ginna Safeguards Contingency Plan
2. ER-SEC.1 Response to Change in Security Threat Level
3. ER-SEC.2 Response to Intrusion by Adversary
4. ER-SEC.3 Response to Airborne Threat
5. NEI 99-01 HU4

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

**Initiating Condition:** Hostile action within the Owner Controlled Area or airborne attack threat

**EAL:**

| | |
|---|---|
| **HA4.1** | **Alert** |
| A hostile action is occurring or has occurred within the Owner Controlled Area as reported by Security Shift Supervision<br>**OR**<br>A validated notification from NRC of an airliner attack threat within 30 min. of the site | |

**Mode Applicability:**

All

**Basis:**

Generic

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

~~These~~ This EAL~~s~~ address<u>es</u> the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

~~EAL #1~~First Condition

This ~~EAL~~ condition addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Owner Controlled Area. Those events are adequately addressed by other EALs.

Note that this ~~EAL~~ condition is applicable for any hostile action occurring, or that has occurred, in the Owner Controlled Area. ~~This includes ISFSI's that may be outside the PROTECTED AREA but still within the OWNER CONTROLLED AREA.~~

~~[Although nuclear plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for OROs to be notified and encouraged to begin activation (if they do not normally) to be better prepared should it be necessary to consider further actions.]~~

~~[If not previously notified by the NRC that the airborne HOSTILE ACTION was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.]~~

~~EAL #2~~Second Condition

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

The intent of this ~~EAL~~ condition is to ensure that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organizations (OROs) and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This ~~EAL~~ condition is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Plant-Specific

Definitions:

**Airliner/Large Aircraft**

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

**Hostile Action**

An act toward Ginna 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 Ginna. 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).

**Ginna Basis Reference(s):**

1. Ginna Safeguards Contingency Plan
2. NEI 99-01 HA4

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

**Initiating Condition:** Hostile action within the Protected Area

**EAL:**

| | |
|---|---|
| **HS4.1** | **Site Area Emergency** |
| A hostile action is occurring or has occurred within the Protected Area as reported by Security Shift Supervision | |

**Mode Applicability:**

All

**Basis:**

Generic

This condition represents an escalated threat to plant safety above that contained in the Alert in that a hostile force has progressed from the Owner Controlled Area to the Protected Area.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires Offsite Response Organization (ORO) readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Protected Area. Those events are adequately addressed by other EALs.

[~~*Although nuclear plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for OROs to be notified and encouraged to begin preparations for public protective actions (if they do not normally) to be better prepared should it be necessary to consider further actions.*~~]

[~~*If not previously notified by NRC that the airborne HOSTILE ACTION was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.*~~]

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

Plant-Specific

None

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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**

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

**Ginna Basis Reference(s):**

1. Ginna Safeguards Contingency Plan
2. NEI 99-01 HS4

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

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

**EAL:**

| **HG4.1** | **General Emergency** |
|---|---|
| A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions | |

**Mode Applicability:**

All

**Basis:**

Generic

~~EAL #1~~

This EAL encompasses conditions under which a hostile action has resulted in a loss of physical control of Vital Areas (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

~~[*Typically, these safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink) for a BWR. The equivalent functions for a PWR are reactivity control, RCS inventory, and secondary heat removal.*]~~

~~[*Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown capability and the location of the transfer switches should be taken into account. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.*]~~

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

~~EAL #2~~

~~This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool.~~

~~[*A freshly off-loaded reactor core is defined by site specific criteria.*]~~

Plant-Specific

Safety functions include:

- Reactivity control
- RCS Inventory
- Secondary Heat Removal

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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).

**Ginna Basis Reference(s):**

1. NEI 99-01 HG1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

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

**EAL:**

**HG4.2 General Emergency**

A hostile action has caused failure of Spent Fuel Cooling systems

**AND**

Imminent fuel damage is likely

**Mode Applicability:**

All

**Basis:**

Generic

~~EAL #1~~

~~This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.~~

~~[*Typically, these safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink) for a BWR. The equivalent functions for a PWR are reactivity control, RCS inventory, and secondary heat removal.*]~~

~~[*Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown capability and the location of the transfer switches should be taken into account. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.*]~~

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

~~EAL #2~~

This EAL addresses failure of spent fuel cooling systems as a result of hostile action if imminent fuel damage is likely~~, such as when a freshly off-loaded reactor core is in the spent fuel pool~~.

~~[*A freshly off-loaded reactor core is defined by site specific criteria.*]~~

Plant-Specific

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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**

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

**Ginna Basis Reference(s):**

1. NEI 99-01 HG1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 5 – Control Room Evacuation

**Initiating Condition:** Control Room evacuation has been initiated

**EAL:**

| **HA5.1** | **Alert** |
| --- | --- |
| Control Room evacuation has been initiated | |

**Mode Applicability:**

All

**Basis:**

Generic

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

Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

Plant-Specific

AP-CR.1 Control Room Inaccessibility provides specific instructions for evacuating the Control Room and establishing plant control in alternate locations (ref. 1).

**Ginna Basis Reference(s):**

1. AP-CR.1 Control Room Inaccessibility
2. NEI 99-01 HA5

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 5 – Control Room Evacuation

**Initiating Condition:** Control Room evacuation has been initiated and plant control **cannot** be established

**EAL:**

| | |
|---|---|
| **HS5.1** | **Site Area Emergency** |
| Control Room evacuation has been initiated | |
| **AND** | |
| Control of the plant **cannot** be established within 30 min. | |

**Mode Applicability:**

All

**Basis:**

Generic

The intent of this ~~IC~~ EAL is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated).

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink)~~ for a BWR. The equivalent functions for a PWR are reactivity control, RCS inventory, and secondary heat removal~~.

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

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

Escalation of this emergency classification level, if appropriate, would be by EALs in Category F or Category R~~Fission Product Barrier Degradation or Abnormal Rad Levels/Radiological Effluent EALs~~.

Plant-Specific

AP-CR.1 Control Room Inaccessibility provides specific instructions for evacuating the Control Room and establishing plant control in alternate locations (ref. 1).

An analysis was performed as part of the Fire Protection Program (ref. 2) to determine how quickly control must be re-established at Ginna without core uncovery or damage. There are 5 time-critical actions which must be accomplished to enable established performance goals to be met. In evaluating a reasonable timeline for completion of tasks required in the ER-FIRE procedures to restore charging, it was estimated that restoration should be completed in less than 30 minutes. This is consistent with information obtained during operator walk-throughs of the ER-FIRE procedures which consistently indicated restoration in 17 to 24 minutes.

**Ginna Basis Reference(s):**

1. AP-CR.1 Control Room Inaccessibility
2. Fire Protection Program, Section 3.2.2.12 Time Criteria for Achieving Hot Shutdown
3. NEI 99-01 HS2

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 6 – Judgment

**Initiating Condition:** *Other conditions existing that in the judgment of the Emergency Director warrant declaration of a UE*

**EAL:**

| **HU6.1** | **Unusual Event** |
|---|---|
| *Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant **OR** indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs* | |

**Mode Applicability:**

All

**Basis:**

<u>Generic</u>

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the ~~NO~~UE emergency classification level.

<u>Plant-Specific</u>

None

**Ginna Basis Reference(s):**

1. NEI 99-01 HU5

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions exist that in the judgment of the Emergency Director warrant declaration of an Alert

**EAL:**

**HA6.1 Alert**

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel **OR** damage to site equipment because of hostile action. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE)

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency classification level.

Plant-Specific

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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).

**Ginna Basis Reference(s):**

1. NEI 99-01 HA6

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions existing that in the judgment of the Emergency Director warrant declaration of a Site Area Emergency

**EAL:**

| **HS6.1** | **Site Area Emergency** |
|---|---|
| Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public **OR** hostile action that results in intentional damage or malicious acts: (1) toward site personnel or equipment that could lead to the likely failure of, or: (2) that prevent effective access to, equipment needed for the protection of the public. **Any** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) beyond the site boundary | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for Site Area Emergency.

Plant-Specific

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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**

The Site Boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. NEI 99-01 HS3

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions exist that in the judgment of the Emergency Director warrant declaration of a General Emergency

**EAL:**

**HG6.1 General Emergency**

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity **OR** hostile action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) offsite for more than the immediate site area

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for General Emergency.

Plant-Specific

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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**

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

**Ginna Basis Reference(s):**

1. NEI 99-01 HG2

**Category S – System Malfunction**

EAL Group: Hot Conditions (RCS 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 AC 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 onsite and offsite power sources for the 480V safeguard buses.

2. Loss of 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 the 125 VDC buses.

3. Criticality & RPS Failure

Inadvertent criticalities pose potential personnel safety hazards as well as being indicative of losses of reactivity control.

Events related to failure of the Reactor Protection System (RPS) to initiate and complete automatic 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 automatic 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, RCS and Containment integrity.

4. Inability to Reach or Maintain Shutdown Conditions

System malfunctions may lead to failure of the plant to be brought to the required plant operating condition required by Technical Specifications if a limiting condition for operation (LCO) is not met.

5. Instrumentation

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of annunciators and indicators are in this subcategory.

6. Communications

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

7. Fuel Clad Degradation

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 Category F, Fission Product Barrier Degradation. 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 and/or the Letdown radiation monitor.

8. RCS Leakage

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor 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 RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Containment integrity.

**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** Loss of **all** offsite AC power to 480V safeguards buses for ≥ 15 min.

**EAL:**

| **SU1.1** **Unusual Event** |
|---|
| Loss of **all** offsite AC power, Table S-1, to 480V safeguards buses for ≥ 15 min. (Note 4) |

Note 4: The ED 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.

| Table S-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

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

**Basis:**

Generic

Prolonged loss of off-site AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power to emergency busses.

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

~~[At multi-unit stations, the EALs should allow credit for operation of installed design features, such as cross-ties or swing diesels, provided that abnormal or emergency operating procedures address their use. However, these stations must also consider the impact of this condition on other shared safety functions in developing the site specific EAL.]~~

~~[Plants that have a proceduralized capability to cross-tie AC power from an off-site power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC.]~~

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., SI pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are not listed in Table S-1 because the availability of power to Buses 17 and 18 alone does not ensure engineered safety features required for hot shutdown and hot standby modes will be operable.

There are three offsite power sources available to these buses (ref. 1):

- *Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)*
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and an Unusual Event must be declared.

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

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. NEI 99-01 SU1

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**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** AC power capability to 480V safeguards buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 480V safeguards bus power

**EAL:**

| | |
|---|---|
| **SA1.1** | **Alert** |
| AC power capability to 480V safeguards buses reduced to a single power source, Table S-1, for ≥ 15 min. (Note 4) | |
| **AND** | |
| **Any** additional single power source failure will result in a complete loss of all 480V safeguards bus power | |

Note 4: The ED 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.

| Table S-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

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

**Basis:**

Generic

~~[This IC and the associated EALs are intended to provide an escalation from IC SU1, "Loss of All Off-site AC Power To Emergency Busses for Greater Than 15 Minutes."]~~

The condition indicated by this ~~IC~~ EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 480V vital bus AC power ~~station blackout~~. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all off-site power and loss of on-site emergency generators with only one train of ~~emergency~~ 480V vital busses being backfed from the unit main generator, or the loss of on-site emergency generators with only one train of ~~emergency~~ 480V vital busses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with EAL SS1.1.

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

~~[At multi-unit stations, the EALs should allow credit for operation of installed design features, such as cross-ties or swing diesels, provided that abnormal or emergency operating procedures address their use. However, these stations must also consider the impact of this condition on other shared safety functions in developing the site specific EAL.]~~

~~[Plants that have a proceduralized capability to cross-tie AC power from an off-site power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC.]~~

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., SI pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are not listed in Table S-1 because the availability of power to Buses 17 and 18 alone does not ensure engineered safety features required for hot shutdown and hot standby modes will be operable.

There are three offsite power sources available to these buses (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and an Unusual Event must be declared.

There are two onsite emergency AC power sources available in the hot modes:

- EDG 1A
- EDG 1B

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to be capable of supplying one or more safety-related buses within 15 minutes, an Unusual Event is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to a Site Area Emergency under EAL SS1.1.

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. NEI 99-01 SA5

**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** Loss of **all** offsite and **all** onsite AC power to 480V safeguards buses for ≥ 15 min.

**EAL:**

| **SS1.1** **Site Area Emergency** |
|---|
| Loss of **all** offsite and **all** onsite AC power, Table S-1, to 480V safeguards buses for ≥ 15 min. (Note 4) |

Note 4: The ED 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.

| Table S-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

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

**Basis:**

Generic

Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to ~~emergency~~ 480V vital busses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency.

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

~~[At multi-unit stations, the EALs should allow credit for operation of installed design features, such as cross-ties or swing diesels, provided that abnormal or emergency operating procedures address~~

~~*their use. However, these stations must also consider the impact of this condition on other shared safety functions in developing the site specific EAL.*~~]

[~~*Plants that have a proceduralized capability to cross-tie AC power from an off site power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC.*~~]

Escalation to General Emergency is via ~~Fission Product Barrier Degradation~~EALs in Category F or ~~IC SG1, "Prolonged Loss of All Off-site Power and Prolonged Loss of All On-site AC Power.~~EAL SG1.1.~~"~~

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., SI pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are not listed in Table S-1 because the availability of power to Buses 17 and 18 alone does not ensure engineered safety features required for hot shutdown and hot standby modes will be operable.

There are three offsite power sources available to these buses in the cold modes (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and Alert must be declared.

There are two onsite emergency AC power sources available in the cold modes:

- EDG 1A
- EDG 1B

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If all sources fail to be capable of supplying all safety-related buses within 15 minutes, a Site Area Emergency is declared under this EAL.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to safeguards buses.

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. ECA-0.0 Loss of All AC Power
3. NEI 99-01 SS1

**Category:** S –System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** Prolonged loss of **all** offsite and **all** onsite AC power to 480V safeguards buses

**EAL:**

**SG1.1 General Emergency**

Loss of **all** offsite and **all** onsite AC power, Table S-1, to 480V safeguards buses

**AND EITHER:**

Restoration of at least one 480V safeguards bus within 4 hours is **not** likely

**OR**

ORANGE or RED path condition exists F-0.2 Core Cooling

| Table S-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

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

**Basis:**

Generic

Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency busses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency.

~~[The (site-specific hours) to restore AC power can be based on a site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout," as available. Appropriate allowance for off site emergency response including evacuation of surrounding areas should be considered. Although this IC may be viewed as redundant to the~~

~~Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.~~]

This ~~IC~~ EAL is specified to assure that in the unlikely event of a prolonged ~~station blackout~~loss of all AC power to 480V safeguards buses, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one ~~emergency~~ safeguards bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded.

~~[Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:~~

~~1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is IMMINENT?~~

~~2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?~~

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

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., SI pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are not listed in Table S-1 because the availability of power to Buses 17 and 18 alone does not ensure engineered safety features required for hot shutdown and hot standby modes will be operable.

There are three offsite power sources available to these buses in the cold modes (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

There are two onsite emergency AC power sources available in the cold modes:

- EDG 1A
- EDG 1B

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to safeguards buses.

Ginna is licensed for a four hour Station Black Out (SBO) coping category (ref. 2). The ability of the plant to cope with a four hour SBO duration was based on an assessment of condensate inventory required for decay heat removal, Class 1E battery capacity, compressed air availability or manual operation of certain valves, effects of loss of ventilation, containment isolation valve operability, and reactor coolant inventory loss. A plant-specific analysis indicates that the expected rates of reactor coolant inventory loss under SBO conditions do not result in core uncovery in a SBO for four hours. Therefore, makeup systems in addition to those currently available under SBO conditions are not required to maintain core cooling under natural circulation. Thus, conditions in which restoration of AC power within four hours is not likely are included in the EAL.

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

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

Thus, indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on ED 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 the existence of conditions to Critical Safety Function Status Tree Core Cooling-ORANGE or RED paths (ref. 3).

**Ginna Basis Reference(s):**

1. UFSAR Section 8 Electrical Power and Figure 8.1-1 Electrical Distribution System
2. Ginna Station Blackout Program Section 3.7
3. CSFST for F-0.2 Core Cooling
4. NEI 99-01 SG1

**Category:** S – System Malfunction

**Subcategory:** 2 – Loss of DC Power

**Initiating Condition:** Loss of **all** vital DC power for ≥ 15 min.

**EAL:**

| | |
|---|---|
| **SS2.1** | **Site Area Emergency** |
| < 108 VDC on **both** 125 VDC buses 1A and 1B for ≥ 15 min. (Note 4) | |

Note 4: The ED 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.

**Mode Applicability:**

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

**Basis:**

Generic

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

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

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

Escalation to a General Emergency would occur by EALs in Category R and Category F~~Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation~~.

Plant-Specific

The 125 VDC vital system is divided into two independent and isolated channels. Each channel consists of one battery, two battery chargers, one DC bus and one inverter. Each inverter has an associated vital AC distribution panel board. Power to the DC bus, DC unit control panels, and inverters is supplied by the station batteries and/or the battery chargers. Each battery charger is fully rated and can recharge a discharged battery while at the same time supplying the steady state power requirements of the system.

A separate TSC Battery system is designed with an intertie to each of the two main (A and B) distribution panels for use during maintenance and abnormal conditions.

The safety-related station batteries have been sized to carry their expected shutdown loads following a plant trip and loss of offsite power or following a station blackout without battery terminal voltage falling below 108.6 volts for a period of 4 hours (ref. 1).

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

The loss of the TSC Battery does not constitute an entry condition for this EAL.

This EAL is the hot condition equivalent of the cold condition loss of DC power EAL CU2.1.

**Ginna Basis Reference(s):**

1. UFSAR Section 8.3.2 Direct Current Power Systems
2. NEI 99-01 SS3

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**Category:** S – System Malfunction

**Subcategory:** 3 – Criticality & RPS Failure

**Initiating Condition:** Inadvertent criticality

**EAL:**

| **SU3.1** | **Unusual Event** |
|---|---|
| An unplanned sustained positive startup rate observed on nuclear instrumentation | |

**Mode Applicability:**

3 - Hot Shutdown, 4 - Hot Standby

**Basis:**

Generic

This ~~IC~~ EAL addresses inadvertent criticality events. While the primary concern of this ~~IC~~ EAL is criticality This ~~IC~~ EAL addresses inadvertent criticality events. This ~~IC~~ EAL indicates a potential degradation of the level of safety of the plant, warranting a ~~NO~~UE classification. This ~~IC~~ EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

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

Escalation would be by ~~the Fission Product Barrier Table~~EALs in Category F, as appropriate to the operating mode at the time of the event.

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the rise in neutron population due to subcritical multiplication. Short-term positive startup rates can also be due to welding activities.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. NEI 99-01 SU8

**Category:** S – System Malfunction

**Subcategory:** 3 – Criticality & RPS Failure

**Initiating Condition:** Automatic trip failed to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

**EAL:**

| | |
|---|---|
| **SA3.1** | **Alert** |

An automatic trip failed to shut down the reactor

**AND**

Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power ≤ 5%

**Mode Applicability:**

1 - Power Operation

**Basis:**

Generic

[The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (~~typically 3 to~~ 5% power). ~~*For plants using CSFSTs, this EAL equates to the criteria used to determine a valid Subcriticality Rod Path. For BWRs this EAL should be the APRM downscale trip setpoint.*~~]

Manual ~~scram (~~trip~~)~~ actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

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

This condition indicates failure of the automatic protection system to ~~scram (~~trip~~)~~ the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a ~~plant transient~~trip signal. Thus the plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad barrier or RCS barrier and because of the failure of the Reactor Protection System to automatically shut down the plant.

If manual actions taken at the reactor control console fail to shut down the reactor, the event would escalate to a Site Area Emergency.

Plant-Specific

A reactor trip is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. The symptoms that require an automatic reactor trip are defined in procedure P-1(ref. 4):

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. 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 negative startup rate as nuclear power drops into the source range (ref. 2).

The operator ensures that the reactor has tripped by (ref. 1):

- Verifying that at least one train of reactor trip breakers are open
- Checking that all control rod position rod bottom lights are on
- Observing neutron flux is decreasing

If these responses cannot be verified, operators perform contingency actions that manually insert control rods, open the reactor trip breakers, and tripping the Rod Drive MG sets. Local opening of these breakers requires actions outside of the Control Room; rapid control rod insertion by these methods is therefore not considered a "successful" manual reactor trip. For purposes of emergency classification, a "successful" manual reactor trip, therefore, includes only those immediate actions taken by the reactor operator in the Control Room to actuate reactor trip switches or deenergize 480 V buses 13 and 15 (ref. 1, 2).

In the event that the operator identifies a reactor trip is imminent and successfully initiates a manual reactor trip before the automatic trip setpoint is reached, no declaration is required. 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. If manual reactor trip actions in the Control Room fail to reduce reactor power below 5% (ref. 3), the event escalates to the Site Area Emergency under EAL SS2.1.

**Ginna Basis Reference(s):**

1. E-0 Reactor Trip or Safety Injection
2. FR-S.1 Response to Reactor Restart/ATWS
3. CSFST for F-0.1 Subcriticality
4. P-1 Reactor Control and Protection System
5. NEI 99-01 SA2

**Category:** S – System Malfunction

**Subcategory:** 3 – Criticality & RPS Failure

**Initiating Condition:** Automatic trip and manual actions taken from the reactor control console failed to shut down the reactor

**EAL:**

**SS3.1 Site Area Emergency**

An automatic trip failed to shut down the reactor as indicated by reactor power > 5%

**AND**

Manual actions taken at the reactor control console failed to shut down the reactor as indicated by reactor power > 5%

**Mode Applicability:**

1 - Power Operation

**Basis:**

Generic

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS.

[The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (~~typically 3 to~~ 5% power). *~~For plants using CSFSTs, this EAL equates to the criteria used to determine a valid Subcriticality Rod Path. For BWRs this EAL should be the APRM downscale trip setpoint.~~*]

Manual scram (trip) actions taken at the reactor control console are any set of actions by the reactor operator(s) ~~at~~ which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual ~~scram (~~trip~~)~~ actions are not considered successful if action away from the reactor control console is required to ~~scram (~~trip~~)~~ the reactor. This EAL is still applicable even if actions taken away from the reactor control console are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant.

[*~~Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.~~*]

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.

Plant-Specific

A reactor trip is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. The symptoms that require an automatic reactor trip are defined in procedure P-1 (ref. 4). Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. 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 negative startup rate as nuclear power drops into the source range (ref. 2).

The operator ensures that the reactor has tripped by (ref. 1):

- Verifying that at least one train of reactor trip breakers are open
- Checking that all control rod position rod bottom lights are on
- Observing neutron flux is decreasing

If these responses cannot be verified, operators perform contingency actions that manually insert control rods, open the reactor trip breakers, and tripping the Rod Drive MG sets. Local opening of these breakers requires actions outside of the Control Room; rapid control rod insertion by these methods is therefore not considered a "successful" manual reactor trip. For purposes of emergency classification, a "successful" manual reactor trip, therefore, includes only those immediate actions taken by the reactor operator in the Control Room to actuate reactor trip switches or deenergize 480V buses 13 and 15 (ref. 1, 2).

If reactor power is above 5%, the reactor is producing more heat than the maximum decay heat load safety systems are designed to remove (ref. 3). Emergency boration is thus required and there is an actual major failure of a system intended for protection of the public. The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Clad and RCS barriers and warrants declaration of a Site Area Emergency.

**Ginna Basis Reference(s):**

1. E-0 Reactor Trip or Safety Injection
2. FR-S.1 Response to Reactor Restart/ATWS
3. CSFST for F-0.1 Subcriticality
4. Procedure P-1 Reactor Control and Protection System
5. NEI 99-01 SS2

**Category:** S – System Malfunction

**Subcategory:** 3 – Criticality & RPS Failure

**Initiating Condition:** Automatic trip and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists

**EAL:**

| | |
|---|---|
| **SG3.1** | **General Emergency** |

An automatic trip failed to shut down the reactor as indicated by reactor power > 5%

**AND**

**All** manual actions fail to shut down the reactor as indicated by reactor power > 5%

**AND EITHER** of the following exist or have occurred:

RED path condition exists F-0.2 Core Cooling

**OR**

RED path condition exists F-0.3 Heat Sink

**Mode Applicability:**

1 - Power Operation

**Basis:**

Generic

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

[The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (~~typically 3 to~~ 5% power). ~~*For plants using CSFSTs, this EAL equates to the criteria used to determine a valid Subcriticality Red Path. For BWRs this EAL should be the APRM downscale trip setpoint.*~~]

[~~*For PWRs, the extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F or that the reactor vessel water level is below the top of active fuel. For plants using CSFSTs, this EAL equates to a Core Cooling RED condition combined with a Subcriticality RED condition.*~~]

[~~*For BWRs, the extreme challenge to the ability to cool the core is intended to mean that the reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level as described in the EOP bases.*~~]

[~~*Another consideration is the inability to initially remove heat during the early stages of this sequence. For PWRs, if emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered*~~

~~to exist. For plants using CSFSTs, this EAL equates to a Heat Sink RED condition combined with a Subcriticality RED condition.~~]

[~~For BWRs, considerations include inability to remove heat via the main condenser, or via the suppression pool or torus (e.g., due to high pool water temperature).~~]

In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier table declaration to permit maximum off-site intervention time.

Plant-Specific

A reactor trip is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. The symptoms that require an automatic reactor trip are defined in procedure P-1 (ref. 6).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. 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 negative startup rate as nuclear power drops into the source range (ref. 2).

The operator ensures that the reactor has tripped by (ref. 1):

- Verifying that at least one train of reactor trip breakers are open
- Checking that all control rod position rod bottom lights are on
- Observing neutron flux is decreasing

If these responses cannot be verified, operators perform contingency actions that manually insert control rods, open the reactor trip breakers, and tripping the Rod Drive MG sets. Local opening of these breakers requires actions outside of the Control Room; rapid control rod insertion by these methods is therefore not considered a "successful" manual reactor trip. For purposes of emergency classification, a "successful" manual reactor trip, therefore, includes only those immediate actions taken by the reactor operator in the

Control Room to actuate reactor trip switches or deenergize 480V buses 13 and 15 (ref. 1, 2).

If reactor power is above 5%, the reactor is producing more heat than the maximum decay heat load safety systems are designed to remove (ref. 3). Emergency boration is thus required and there is an actual major failure of a system intended for protection of the public. The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Clad and RCS barriers.

CSFST Core Cooling RED path condition represents a severe challenge to the core cooling function (ref. 4). Core Exit Thermocouples (CETs) are an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. RCS temperatures > 1200 °F or > 700 °F with reactor vessel water level below the top of active fuel signals the transition from a subcooled to a superheated regime. In a superheated regime, heat transfer mechanics are not as efficient as the subcooled condition and could lead to film boiling and a rapid rise in clad temperatures. This condition is considered a Fuel Clad barrier loss condition because the possible rapid rise in clad temperatures may lead to clad failure.

CSFST Heat Sink RED path condition represents a severe challenge to the heat removal function (ref. 5). Inability to remove heat from the RCS to the ultimate heat sink (lake or atmosphere) is a loss of function required for hot shutdown with the reactor at pressure and temperature and thus represents potential loss of the Fuel Clad and RCS barriers. Heat Sink RED path conditions are based on a combination of inadequate S/G level (< 5%) and inadequate feedwater flow (< 200 gpm total S/G feedwater flow).

The combination of these conditions (reactor power greater than 5% with loss of core cooling or inability to remove heat from the RCS) indicates the ultimate heat sink function is under extreme challenge, a core melt sequence may exist and rapid degradation of the fuel clad could begin. To permit maximum offsite intervention time, the General Emergency declaration is appropriate in anticipation of an inevitable General Emergency declaration due to loss and potential loss of fission product barriers.

**Ginna Basis Reference(s):**

1. E-0 Reactor Trip or Safety Injection
2. FR-S.1 Response to Reactor Restart/ATWS
3. CSFST for F-0.1 Subcriticality
4. CSFST for F-0.2 Core Cooling
5. CSFST for F-0.3 Heat Sink
6. P-1 Reactor Control and Protection System
7. NEI 99-01 SG2

**Category:** S – System Malfunction

**Subcategory:** 4 – Inability to Reach or Maintain Shutdown Conditions

**Initiating Condition:** Inability to reach required shutdown within Technical Specification limits

**EAL:**

| | |
|---|---|
| **SU4.1** | **Unusual Event** |
| Plant is **not** brought to required operating mode within Technical Specifications LCO required action completion time | |

**Mode Applicability:**

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

**Basis:**

Generic

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a four hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable required action ~~statement~~ completion time in the Technical Specifications. An immediate N~~O~~UE is required when the plant is not brought to the required operating mode within the allowable required action ~~statement~~ completion time in the Technical Specifications. Declaration of a N~~O~~UE is based on the time at which the LCO-specified required action ~~statement~~ completion time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

[~~*Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.*~~]

Plant-Specific

None

**Ginna Basis Reference(s):**

1. Technical Specifications 3.0, Limiting Conditions for Operations (LCO) Applicability
2. NEI 99-01 SU2

**Category:** S – System Malfunction

**Subcategory:** 5 – Instrumentation

**Initiating Condition:** Unplanned loss of safety system annunciation or indication in the Control Room for ≥ 15 min.

**EAL:**

| **SU5.1** | **Unusual Event** |
|---|---|
| Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4) | |

Note 4: The ED 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

| **Table S-2 Vital Control Room Panels** | | | | | | | |
|---|---|---|---|---|---|---|---|
| A | AA | B | C | D | E | F | G |

**Mode Applicability:**

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

**Basis:**

Generic

This ~~IC and its associated~~ EAL ~~are~~ is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered ~~[e.g., SPDS, plant computer, etc.]~~.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that ~~most~~ plant designs provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10CFR50.72. If the shutdown is not in compliance with the Technical Specification action, the ~~NO~~UE is based on EAL SU~~2 "Inability to Reach Required Shutdown Within Technical Specification Limits."~~4.1.

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

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

~~[Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no IC is indicated during these modes of operation.]~~

This ~~NO~~UE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a significant transient is in progress during the loss of annunciation or indication.

Plant-Specific

Control Room Panels A through G, Table S-2, provide safety-related indications and annunciation in the Main Control Room (ref. 1, 2).

A 75% loss of annunciators is defined as loss of 6 of 8 annunciator panels listed on Table S-2. Loss of 75% of MCB indications is loss of 75% of the indications on the center and left sections of the main control board indications.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. UFSAR Sections 7.5 Safety-Related Display Instrumentation
2. ER-INST.2 Loss of Annunciators
3. NEI 99-01 SU3

**Category:** S – System Malfunction

**Subcategory:** 5 – Instrumentation

**Initiating Condition:** Unplanned loss of safety system annunciation or indication in the Control Room with either (1) a significant transient in progress, or (2) compensatory indicators are unavailable

**EAL:**

**SA5.1 Alert**

Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)

**AND EITHER:**

A significant transient is in progress, Table S-3

**OR**

Compensatory indications are unavailable (PPCS)

Note 4: The ED 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.

| Table S-2 Vital Control Room Panels | | | | | | | |
|---|---|---|---|---|---|---|---|
| A | AA | B | C | D | E | F | G |

| Table S-3 Significant Transients |
|---|
| • Automatic turbine runback > 25% thermal power<br>• Electric load rejection > 25% full electrical load<br>• Reactor trip<br>• Safety Injection activation |

**Mode Applicability:**

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

**Basis:**

Generic

This ~~IC~~ EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a significant transient.

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

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift ~~Supervisor~~ Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the ~~N~~OUE is based on EAL SU~~2 "Inability to Reach Required Shutdown Within Technical Specification Limits."~~4.1.

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

"Compensatory indications" in this context includes computer based information such as Plant Process Computer System ~~and SPDS~~. ~~[This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.] If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.~~

~~[Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no IC is indicated during these modes of operation.]~~

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

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a significant transient in progress during the loss of annunciation or indication.

Plant-Specific

Control Room Panels A through G, Table S-2, provide safety-related indications and annunciation in the Main Control Room (ref. 1, 2).

PPCS is considered compensatory indication.

Significant transients are listed in Table S-3.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. UFSAR Sections 7.5 Safety-Related Display Instrumentation
2. ER-INST.2 Loss of Annunciators
3. NEI 99-01 SA4

**Category:** S – System Malfunction
**Subcategory:** 5 – Instrumentation
**Initiating Condition:** Inability to monitor a significant transient in progress
**EAL:**

**SS5.1 Site Area Emergency**

Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)

**AND**

A significant transient is in progress, Table S-3

**AND**

Compensatory indications are unavailable (PPCS)

Note 4: The ED 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

**Table S-2 Vital Control Room Panels**

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| A | AA | B | C | D | E | F | G |

**Table S-3 Significant Transients**

- Automatic turbine runback > 25% thermal power
- Electric load rejection > 25% full electrical load
- Reactor trip
- Safety Injection activation

**Mode Applicability:**

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

**Basis:**

Generic

This ~~IC~~ EAL is intended to recognize the threat to plant safety associated with the complete loss of capability of the control room staff to monitor plant response to a significant transient.

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

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift ~~Supervisor~~ Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. *This will be addressed by the specific Technical Specification. The initiation of a Technical* Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NOUE is based on EAL SU~~2 "Inability to Reach Required Shutdown Within Technical Specification Limits."~~4.1

A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

~~[Site specific a~~Annunciators for this EAL ~~should be~~are limited to include those identified in the Abnormal Operating Procedures~~,~~ and in the Emergency Operating Procedures, and in other EALs (e. g., area, process, and/or effluent rad monitors, etc.)]

~~Site specific i~~Indications needed to monitor safety functions necessary for protection of the public ~~must~~ include control room indications, computer generated indications and dedicated annunciation capability.

~~[*The specific indications should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, maintain the spent fuel cooled, and to maintain containment intact.*]~~

"Compensatory indications" in this context includes computer based information such as Plant Process Computer System~~ and SPDS~~. ~~This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.~~

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

~~[*Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no IC is indicated during these modes of operation.*]~~

Plant-Specific

Control Room Panels A through G, Table S-2, provide safety-related indications and annunciation in the Main Control Room (ref. 1, 2).

PPCS is considered compensatory indication.

Significant transients are listed in Table S-3.

**Ginna Basis Reference(s):**

1. UFSAR Sections 7.5 Safety-Related Display Instrumentation
2. ER-INST.2 Loss of Annunciators
3. NEI 99-01 SS6

**Category:** S – System Malfunction
**Subcategory:** 6 – Communications
**Initiating Condition:** Loss of **all** onsite or offsite communications capabilities
**EAL:**

**SU6.1 Unusual Event**

Loss of **all** Table S-4 onsite (internal) communication methods affecting the ability to perform routine operations

**OR**

Loss of **all** Table S-4 offsite (external) communication methods affecting the ability to perform offsite notifications

**Table S-4 Communications Systems**

| System | Onsite (internal) | Offsite (external) |
|---|---|---|
| Commercial phone system | X | X |
| Direct Dial POTS Lines (Blue Phones) | X | X |
| Plant Page Party system | X | |
| Radios/Walkie Talkies | X | |
| FTS 2001 telephone system (ENS, HPN) | | X |
| Control Room Hard Wired Satellite Phone | | X |
| Control Room Emergency Cell Phone | | X |

**Mode Applicability:**

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

**Basis:**

Generic

The purpose of this~~ IC and its associated~~ EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities.

~~[~~The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.~~]~~

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to off-site locations, etc.) are being used to make communications possible.

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

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

<u>Plant-Specific</u>

Onsite/offsite communications systems are listed in Table S-4 (ref. 1, 2).

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

**Ginna Basis Reference(s):**

1. A-56 Communication Systems at Ginna Station
2. ER-COMM.1 Loss of Communications
3. NEI 99-01 SU6

**Category:** S – System Malfunction
**Subcategory:** 7 – Fuel Clad Degradation
**Initiating Condition:** Fuel clad degradation

**EAL:**

| **SU7.1** | **Unusual Event** |
|---|---|
| RCS specific activity > 60 μCi/gm dose equivalent I-131 | |

**Mode Applicability:**

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

**Basis:**

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the ~~Fission Product Barriers~~EALs in Category F.

~~EAL #1~~

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

~~[*Such as BWR air ejector monitors, PWR failed fuel monitors, etc.*]~~

~~EAL #2~~

This threshold addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits.

Plant-Specific

This EAL addresses reactor coolant samples exceeding Technical Specification 3.4.16 which is applicable in Modes 1, 2 and in Mode 3 with RCS average temperature ($T_{avg}$) $\geq$ 500 °F. Though the referenced Technical Specification limits are mode dependent, it is appropriate that the EAL be applicable in all hot modes, as it indicates a potential degradation in the level of safety of the plant. The Technical Specification limits accommodate an iodine spike phenomenon that may occur following changes in thermal power and during reactor startup and shutdown. The Technical Specification LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident (ref. 1).

**Ginna Basis Reference(s):**

1. Technical Specification 3.4.16 Reactor Coolant System - RCS Specific Activity
2. NEI 99-01 SU4

**Category:** S – System Malfunction

**Subcategory:** 7 – Fuel Clad Degradation

**Initiating Condition:** Fuel clad degradation

**EAL:**

| **SU7.2** | **Unusual Event** |
|---|---|
| Valid Letdown Monitor (R-9) reading ≥ 4800 mR/hr | |

**Mode Applicability:**

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

**Basis:**

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

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

Plant-Specific

This EAL addresses indication of gross failed fuel that may be in excess of Technical Specification (ref. 1) coolant activity limits.

The Letdown Line Monitor (R-9) gross radiation channel continuously monitors the activity in a sample drawn from the RCS (NaOH tank room) and actuates an alarm in the Control Room if a predetermined activity level is reached. The high alarm setting of 200 mRad/hr ensures timely detection of failed fuel increases greater than 0.1% (ref. 2, 3, 4).

The 4800 mR/hr value for R-9 is based on total RCS activity corresponding to 60 μCi/gm I-131 equivalent and 1% failed fuel (100 / $\bar{E}$). A shielding calculation was performed to obtain this value (ref. 5).

Definitions:

**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.

**Ginna Basis Reference(s):**

1. Technical Specification 3.4.16 Reactor Coolant System - RCS Specific Activity
2. AP-RCS.3 High Reactor Coolant Activity
3. AR-RMS-9 R9 Letdown Line Monitor
4. P-9 Radiation Monitoring System
5. CALC-2011-0019, "R9 Letdown Line Radiation Monitor NEI 99-01 Rev. 5 Evaluation"
6. NEI 99-01 SU4

**Category:** S – System Malfunction
**Subcategory:** 8 – RCS Leakage
**Initiating Condition:** RCS leakage

**EAL:**

**SU8.1 Unusual Event**

Unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 4)

**OR**

Identified leakage > 25 gpm for ≥ 15 min. (Note 4)

Note 4: The ED 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.

**Mode Applicability:**

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

**Basis:**

Generic

This ~~IC~~ EAL is included as a ~~NO~~UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation should be excluded from this ~~IC~~EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this ~~IC~~ EAL if the relief valve cannot be isolated. 15 minutes allows time to evaluate the source and take corrective actions to isolate the leak.

The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this ~~IC~~ EAL to the Alert level is via ~~Fission Product Barrier Degradation ICs~~EALs in Category F.

Plant-Specific

Technical Specifications Section 3.4.13 RCS Operational Leakage prescribes RCS leakage limits for pressure boundary (none allowed), unidentified (1 gpm) and identified (10 gpm) leakage (ref. 1). AP-RCS.1 Reactor Coolant Leak provide direction for determining RCS leakage for off normal events and for operations troubleshooting (ref. 2).

**Ginna Basis Reference(s):**

1. Technical Specifications 3.4.13, RCS Operational Leakage
2. AP-RCS.1 Reactor Coolant Leak
3. NEI 99-01 SU5

**Category F – Fission Product Barrier Degradation**

EAL Group: Hot Conditions (RCS 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. Reactor Fuel Clad (FC): The Fuel Clad barrier consists of the zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
C. Containment (CNMT): 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.

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:

*Unusual Event:*
***ANY** loss or **ANY** potential loss of Containment*
*Alert:*

***ANY** loss or **ANY** potential loss of either Fuel Clad or RCS*

*<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*

The logic used for Category F EALs reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. UE EALs associated with RCS and Fuel Clad Barriers are addressed under Category S.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" thresholds existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" thresholds existed, the ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.
- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier.

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

**Initiating Condition:** **ANY** loss or **ANY** potential loss of Containment

**EAL:**

| | |
|---|---|
| **FU1.1** | **Unusual Event** |
| **ANY** loss or **ANY** potential loss of Containment (Table F-1) | |

**Mode Applicability:**

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

**Basis:**

Generic

None

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Fuel Clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Containment barrier in combination with the loss or potential loss of either the Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

**Ginna Basis Reference(s):**

1. NEI 99-01 FU1

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

**Initiating Condition:** **ANY** loss or **ANY** potential loss of either Fuel Clad or RCS

**EAL:**

| | |
|---|---|
| **FA1.1** | **Alert** |
| **ANY** loss or **ANY** potential loss of either Fuel Clad or RCS (Table F-1) | |

**Mode Applicability:**

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

**Basis:**

Generic

None

Plant-Specific

Fuel Clad, RCS 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 RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

**Ginna 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 Shutdown, 4 - Hot Standby

**Basis:**

Generic

None

Plant-Specific

Fuel Clad, RCS 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 RCS 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 RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less imminent.

**Ginna 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 the third barrier

**EAL:**

| | |
|---|---|
| **FG1.1** | **General Emergency** |
| Loss of **ANY** two barriers<br>**AND**<br>Loss or potential loss of the third barrier (Table F-1) | |

**Mode Applicability:**

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

**Basis:**

Generic

None

Plant-Specific

Fuel Clad, RCS 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, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

**Ginna Basis Reference(s):**

1. NEI 99-01 FG1

# ATTACHMENT 2

# FISSION PRODUCT BARRIER LOSS/POTENTIAL LOSS MATRIX AND BASES

**Introduction**

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that the three barriers occupy 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. CSFSTs

B. Core Exit TCs

C. Inventory

D. Radiation / Coolant Activity

E. Isolation Status

F. Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the category rows and the Loss/Potential Loss columns. The intersection of each category row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A is "FC Loss A.1," the third Containment barrier Potential Loss is "CNMT P-Loss B.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 row of fission product barrier Loss and Potential Loss thresholds in that category to determine if any threshold has been exceeded. If a threshold has not been exceeded in that category row, the EAL-user proceeds to the next likely category and continues review of the row of thresholds in the new category

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 (i.e., greater than 1.0E+03 R/hr), a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier exist. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS 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 Matrix | | | | | | |
|---|---|---|---|---|---|---|
| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Containment Barrier | |
| Category | Loss | Potential Loss | Loss | Potential Loss | Loss | Potential Loss |
| **A** CSFST | 1. RED path condition exists F-0.2 Core Cooling | 1. ORANGE path condition exists F-0.2 Core Cooling<br>2. RED path condition exists F-0.3 Heat Sink and heat sink is required | None | 1. RED path condition exists F-0.4 Integrity<br>2. RED path condition exists F-0.3 Heat Sink and heat sink is required | None | 1. RED path condition exists F-0.5 Containment |
| **B** Core Exit TCs | 2. Core Exit TCs ≥ 1,200°F | 3. Core Exit TCs ≥ 700°F | None | None | None | 2. Core Exit TCs cannot be restored < 1,200°F within 15 min.<br>3. Core Exit TCs ≥ 700°F **AND** RVLIS level cannot be restored > 52% [> 55% adverse CNMT] with no RCPs running within 15 min. |
| **C** Inventory | None | 4. RVLIS level ≤ 52% [≤ 55% adverse CNMT] **OR** At least one RCP running RVLIS fluid fraction ≤ 66% | 1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< EOP Fig. MIN SUBCOOLING)<br>2. Ruptured S/G results in an ECCS (SI) actuation | 3. RCS leak rate > 50 gpm with letdown isolated | 1. A containment pressure rise followed by a rapid unexplained drop in containment pressure<br>2. Containment pressure or sump level response not consistent with LOCA conditions<br>3. Ruptured S/G is also faulted outside of containment<br>4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable prolonged steam release from affected S/G to the environment | 4. Containment pressure ≥ 60 psig and rising<br>5. Containment hydrogen concentration ≥ 4%<br>6. Containment pressure ≥ 28 psig and < two CRFC units and one CS pump operating per design |
| **D** Radiation / Coolant Activity | 3. Containment radiation monitor R-29/R-30 reading > 1.0E+02 R/hr<br>4. Valid Letdown Monitor (R-9) reading ≥ 24,000 mR/hr<br>5. Coolant activity >300 μCi/gm dose equivalent I-131 | None | 3. Containment radiation monitor R-29/R-30 reading > 1.0E+01 R/hr | None | None | 7. Containment radiation monitor R-29/R-30 reading > 1.0E+03 R/hr |
| **E** Isolation Status | None | None | None | None | 5. Failure of all valves in ANY one line to close **AND** Direct downstream pathway to the environment exists after containment isolation signal | None |

**Table F-1 Fission Product Barrier Matrix**

| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Containment Barrier | |
|---|---|---|---|---|---|---|
| Category | Loss | Potential Loss | Loss | Potential Loss | Loss | Potential Loss |
| F Judgment | 6. ANY condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier | 5. ANY condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier | 4. ANY condition in the opinion of the Emergency Director that indicates loss of the RCS barrier | 4. ANY condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier | 6. ANY condition in the opinion of the Emergency Director that indicates loss of the containment barrier | 8. ANY condition in the opinion of the Emergency Director that indicates potential loss of the containment barrier |

**Barrier:** Fuel Clad

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 1. RED path condition exists F-0.2 Core Cooling |

**Basis:**

Generic

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

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path is given in F-0.2 and indicates significant core exit superheating and core uncovery (ref. 1).

RED path conditions exist if either:

- Core Exit TCs are $\geq$ 1200°F
- Core Exit TCs are $\geq$ 700°F with RVLIS $\leq$ 52% [$\leq$ 55% adverse CNMT] with no RCPs running

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 2):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling

**Barrier:** Fuel Clad

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 1. ORANGE path condition exists F-0.2 Core Cooling |

**Basis:**

Generic

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur.

~~Heat Sink – RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.~~

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path is given in F-0.2 and indicates subcooling has been lost and that some fuel clad damage may potentially occur (ref. 1).

ORANGE path Core Cooling conditions exist if, with RCS subcooling < requirements of EOP Fig. MIN SUBCOOLING, either:

- with no RCPs running Core Exit TCs are ≥ 700°F or RVLIS level ≤ 52% [55% adverse CNMT]

  OR

- with at least one RCP running RVLIS fluid fraction ≤ 66%

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 2):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. FR-C.2 Response to Degraded Core Cooling

**Barrier:** Fuel Clad

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 2. RED path condition exists F-0.3 Heat Sink and heat sink is required |

**Basis:**

Generic

~~Core Cooling – ORANGE indicates subcooling has been lost and that some clad damage may occur.~~

Heat Sink - RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.

Plant-Specific

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path in F-0.3 (ref. 1). The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. Procedure FR-H.1, Response to Loss of Secondary Heat Sink, indicates heat sink is required when RCS pressure is greater than any non-faulted SG pressure and RCS cold leg temperature is greater than 350°F (ref. 2).

RED path Heat Sink conditions exist if both of the following:

- Narrow Range level in both S/Gs is ≤ 7% [≤ 25% adverse CNMT]

  AND

- Total feedwater flow to SGs is ≤ 200 gpm

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 3):

- Containment pressure > 4 psig, or

- Containment radiation > $10^5$ R/hr

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

**Ginna Basis Reference(s):**

1. CSFST for F-0.3 Heat Sink
2. FR-H.1 Response to Loss of Secondary Heat Sink
2. FR-C.2 Response to Degraded Core Cooling

**Barrier:** Fuel Clad

**Category:** B. Core Exit TCs

**Degradation Threat:** Loss

**Threshold:**

| 2. Core Exit TCs ≥ 1,200°F |
|---|

**Basis:**

Generic

~~[Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked) or plants which do not have a CSF scheme.]~~

~~Loss Threshold A~~

The 1,200°F~~site specific~~ reading ~~should~~ corresponds to significant superheating of the coolant.

~~[This value typically corresponds to the temperature reading that indicates core cooling – RED in Fuel Clad Barrier loss threshold 1.A which is usually about 1200 degrees F.]~~

~~Potential Loss Threshold A~~

~~The site specific reading should correspond to loss of subcooling.~~

~~[This value typically corresponds to the temperature reading that indicates core cooling - ORANGE in Fuel Clad Barrier potential loss threshold 1.A which is usually about 700 to 900 degrees F.]~~

Plant-Specific

Core Exit Thermocouples (TCs) reading at or in excess of 1200°F corresponds to the CSFST Core Cooling RED path entry condition (ref. 1). Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation (ref. 2). Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery. Events that result in Core Exit TC readings above the loss threshold are classified severe accidents and lead to entry into Severe Accident Management Guidelines (ref. 3).

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. EPIP-2-16 Core Damage Estimation
3. Ginna Severe Accident Management Guidelines
4. FR-C.1 Response To Inadequate Core Cooling

**Barrier:** Fuel Clad

**Category:** B. Core Exit TCs

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 3. Core Exit TCs ≥ 700°F |

**Basis:**

Generic

[~~*Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked) or plants which do not have a CSF scheme.*~~]

~~Loss Threshold A~~

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

[~~*This value typically corresponds to the temperature reading that indicates core cooling - RED in Fuel Clad Barrier loss threshold 1.A which is usually about 1200 degrees F.*~~]

~~Potential Loss Threshold A~~

~~The site specific reading should~~Core Exit TC readings ≥ 700°F correspond to loss of subcooling.

[~~*This value typically corresponds to the temperature reading that indicates core cooling - ORANGE in Fuel Clad Barrier potential loss threshold 1.A which is usually about 700 to 900 degrees F.*~~]

Plant-Specific

Core Exit Thermocouples (TCs) reading at or in excess of 700°F corresponds to the CSFST Core Cooling ORANGE path entry criteria (ref. 1). Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation (ref. 2). RCS superheat, as indicated by Core Exit TCs reading at or in excess of 700°F, signals the transition from a subcooled to a superheated regime. In a superheated regime, heat transfer mechanics are not as efficient as the subcooled condition and could lead to a rapid rise in clad temperatures. Valid indication of superheat is a potential Fuel Clad barrier loss condition because the possible rapid rise in clad temperatures may lead to clad failure.

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. EPIP-2-16 Core Damage Estimation
3. FR-C.1 Response To Inadequate Core Cooling

**Barrier:** Fuel Clad

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 4. RVLIS ≤ 52% [≤ 55% adverse CNMT]<br>**OR**<br>At least one RCP running RVLIS fluid fraction ≤ 66% |

**Basis:**

Generic

There is no Loss threshold associated with this item.

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

[~~For sites using CSFSTs, the Potential Loss~~ threshold ~~is defined by the Core Cooling – ORANGE path. The site specific value in this~~ threshold ~~should be consistent with the CSFST value.~~]

Plant-Specific

The Reactor Vessel water level threshold is used in the EOPs to signal core uncovery and is, therefore, indication of inadequate coolant inventory. If the RVLIS indication drops to 52% [≤ 55% adverse CNMT] **OR** with at least one RCP running RVLIS fluid fraction ≤ 66%, a core covered condition cannot be confirmed. According to the Core Cooling-ORANGE path, this water level indicates subcooling has been lost and that some fuel clad damage may occur. (ref. 1, 2)

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 2):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. FR-C.2 Response to Degraded Core Cooling

**Barrier:** Fuel Clad

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 3. Containment radiation monitor R-29/R-30 reading > 1.0E+02 R/hr |

**Basis:**

Generic

The ~~site specific~~1.0E+02 R/hr containment radiation monitor reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment.

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

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage.

This value is higher than that specified for RCS barrier Loss threshold #3~~6~~. Thus, this threshold indicates a loss of both the Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

~~[**Caution:** it is important to recognize that in the event the radiation monitor is sensitive to shine from the reactor vessel or piping, spurious readings will be present and another indicator of fuel clad damage is necessary or compensated for in the threshold value.]~~

There is no Potential Loss threshold associated with this item.

Plant-Specific

Containment radiation is indicated on R-29 and R-30 (ref. 1).

R-29 & R-30 alert alarms at 1.0E+01 R/hr, indicative of a significant RCS breach (LOCA) in containment (~0.1% gap activity). The R-29 & R-30 high alarm setpoint is set at 1.0E+02 R/hr and is indicative of a significant gap activity release into containment and thus considered a loss of the fuel clad barrier. A reading on containment radiation monitors greater than 1.0E+03 R/hr is indicative of significant fuel activity and thus considered a potential loss of Containment (ref. 2).

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. EPIP-2-16 Core Damage Estimation

**Barrier:** Fuel Clad

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 4. Valid Letdown Line Monitor (R-9) reading ≥ 24,000 mR/hr |

**Basis:**

Generic

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

[~~The value can be expressed either in mR/hr observed on the sample or as µCi/gm results from analysis.~~]

There is no Potential Loss threshold associated with this item.

Plant-Specific

The Letdown Line Monitor (R-9) gross radiation channel continuously monitors the activity in a sample drawn from the RCS (NaOH tank room) and actuates an alarm in the Control Room if a predetermined activity level is reached. A Letdown Line Monitor reading of 24,000 mR/hr represents fuel clad damage of approximately 5% corresponding to the reactor coolant activity fuel Clad loss threshold of 300 µCi/gm dose equivalent I-131 (ref. 2).

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. CALC-2011-0019, R9 Letdown Line Radiation Monitor NEI 99-01 Rev. 5 Evaluation.

**Barrier:** Fuel Clad

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 5. Coolant activity >300 µCi/gm dose equivalent I-131 |

**Basis:**

Generic

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

~~[The value can be expressed either in mR/hr observed on the sample or as µCi/gm results from analysis.]~~

There is no Potential Loss threshold associated with this item.

Plant-Specific

None

**Ginna Basis Reference(s):**

1. NEI 99-01 Revision 5, pg 35

**Barrier:** Fuel Clad

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Potential Loss

**Threshold:**

| None |
| --- |

**Barrier:** Fuel Clad

**Category:** E. Isolation Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** E. Isolation Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier |

**Basis:**

Generic

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

Plant-Specific

The Emergency Director 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.

**Ginna Basis Reference(s):**

None

**Barrier:** Fuel Clad

**Category:** F. Judgment

**Degradation Threat:** Potential Loss

**Threshold:**

5. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier

**Basis:**

Generic

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

Plant-Specific

The Emergency Director 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.

**Ginna Basis Reference(s):**

None

**Barrier:** Reactor Coolant System

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 1. RED path condition exists F-0.4 Integrity |

**Basis:**

Generic

RCS Integrity - RED indicates an extreme challenge to the safety function derived from appropriate instrument readings.

~~Heat Sink - RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.~~

There is no Loss threshold associated with this item.

Plant-Specific

Critical Safety Function Status Tree (CSFST) Integrity-RED path is given in F-0.4 and entry is indicative of a direct threat to RCS integrity due to imminent pressurized thermal shock (ref. 1, 2).

RED path Integrity conditions exist if:

- temperature lowers in either RCS cold leg ≥ 100°F/hr

  AND

- temperature in either RCS cold leg is ≤ 284°F

**Ginna Basis Reference(s):**

1. FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
2. CSFST for F-0.4 Integrity

**Barrier:** Reactor Coolant System

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| **2. RED path condition exists F-0.3 Heat Sink and heat sink is required** |

**Basis:**

Generic

~~RCS Integrity - RED indicates an extreme challenge to the safety function derived from appropriate instrument readings.~~

Heat Sink - RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.

There is no Loss threshold associated with this item.

Plant-Specific

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path in F-0.3 (ref. 1). The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. Procedure FR-H.1, Response to Loss of Secondary Heat Sink, indicates heat sink is required when RCS pressure is greater than any non-faulted SG pressure and RCS cold leg temperature is greater than 350°F (ref. 2).

RED path Heat Sink conditions exist if both of the following:

- Narrow Range level in both S/Gs is ≤ 7% [≤ 25% adverse CNMT]

  AND

- Total feedwater flow to SGs is ≤ 200 gpm

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 3):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

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

**Ginna Basis Reference(s):**

1. FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
2. FR-H.1 Response to Loss of Secondary Heat Sink
3. CSFST for F-0.4 Integrity

**Barrier:** Reactor Coolant System

**Category:** B. Core Exit TCs

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** B. Core Exit TCs

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< EOP Fig. MIN SUBCOOLING)

**Basis:**

Generic

~~Loss Threshold A~~

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

~~Potential Loss Threshold A~~

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

~~[*For plants with low capacity charging pumps, a 50 gpm indicated leak rate value may be used to indicate the Potential Loss.*]~~

Plant-Specific

Critical Safety Function Status Trees (CSFST) Core Cooling indicates that if subcooling margin based on core exit TCs is in the Inadequate Subcooling Region of EOP Fig. MIN SUBCOOLING, a loss of RCS subcooling has occurred (ref. 1, 4). E-0, Reactor Trip or Safety Injection and AP-RCS.1, Reactor Coolant Leak, provide appropriate actions to prevent and mitigate the consequences of RCS leakage (ref. 2, 3).

AP-RCS.1 provides a list of conditions that may be observed when excessive RCS leakage occurs and provides appropriate actions to prevent and mitigate the consequences of RCS leakage (ref. 3).

The loss of subcooling as a result of inability to establish RCS heat transfer to the ultimate heat sink is indicative of potential losses of the Fuel Clad and RCS barriers.

**Ginna Basis Reference(s):**

1. F-0.2 CSFST Core Cooling
2. E-0 Reactor Trip or Safety Injection
3. AP-RCS.1 Reactor Coolant Leak
4. EOP Figure MIN SUBCOOLING
5. AP-CVCS.1 CVCS leak

**Barrier:** Reactor Coolant System

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 2. Ruptured S/G results in an ECCS (SI) actuation |

**Basis:**

Generic

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

~~[For plants that have implemented Westinghouse Owners Group emergency response guides, this condition is described by "entry into E-3 required by EOPs".]~~

By itself, this threshold will result in the declaration of an Alert. However, if the SG is also faulted (*i.e., two barriers failed), the declaration escalates to a Site Area Emergency per* Containment barrier loss thresholds.

There is no potential loss threshold associated with this item.

Plant-Specific

In conjunction with Containment Loss C.3 and the Fuel Clad barrier thresholds, this threshold addresses the full spectrum of Steam Generator Tube Rupture (SGTR) events. A ruptured SG is primary-to-secondary leakage through the steam generator tubes. ECCS (SI) actuation is caused by (ref. 1):

- Containment pressure > 4 psig
- Pressurizer pressure < 1750 psig
- Steam line pressures < 514 psig

Indications of a ruptured S/G include (ref. 2):

- Unexpected rise in either S/G narrow range level
- High radiation on Main Steamline Radiation Monitors

- Local indications of increase steamline radiation

Definitions:

**Ruptured**

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

**Ginna Basis Reference(s):**

1. E-0 Reactor Trip or Safety Injection
2. E-3 Steam Generator Tube Rupture
3. AP-RCS.1 Reactor Coolant Leak

**Barrier:** Reactor Coolant System

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 3. RCS leak rate > 50 gpm with letdown isolated |

**Basis:**

Generic

~~Loss Threshold A~~

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

~~Potential Loss Threshold A~~

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

[~~*For plants with low capacity charging pumps, a 50 gpm indicated leak rate value may be used to indicate the Potential Loss.*~~]

Plant-Specific

The CVCS includes three positive displacement horizontal pumps with a capacity of 46 gpm each (ref. 1). The pressurizer level control program regulates letdown purification subsystem flow by adjusting the letdown flow control valve so that the reactor coolant pump (RCP) controlled leak-off plus the letdown flow matches the input from the operating charging pump. Equilibrium pressurizer level conditions may be disturbed due to RCS temperature changes, power changes, or RCS inventory loss due to leakage. A decrease in pressurizer water level below the programmed level results in a control signal to start one or both standby charging pumps to restore water level. The need for a second or third charging pump to makeup leakage in excess of letdown flow would be indicative of substantial RCS leakage. The single charging pump capacity is rounded up to 50 gpm for this threshold and clearly signals that operation of more than one charging pump is needed (ref. 2).

**Ginna Basis Reference(s):**

1. UFSAR Table 9.3.6 CVCS Performance Parameters
2. UFSAR Section 9.3.4.2.2.2 Charging Pump Control

**Barrier:** Reactor Coolant System

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 3. Containment radiation monitor R-29/R-30 reading > 1.0E+01 R/hr |

**Basis:**

Generic

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

~~[The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the containment atmosphere.]~~

This reading ~~will be~~is less than that specified for Fuel Clad barrier threshold ~~6~~3. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad barrier threshold, fuel damage would also be indicated.

~~[However, if the site specific physical location of the containment radiation monitor is such that radiation from a cloud of released RCS gases could not be distinguished from radiation from adjacent piping and components containing elevated reactor coolant activity, this threshold should be omitted and other site specific indications of RCS leakage substituted.]~~

There is no Potential Loss threshold associated with this item.

Plant-Specific

Containment radiation is indicated on R-29 and R-30 (ref. 1).

R-29 & R-30 alert alarms at 1.0E+01 R/hr, indicative of a significant RCS breach (LOCA) in containment (~0.1% gap activity). The R-29 & R-30 high alarm setpoint is set at 1.0E+02 R/hr and is indicative of a significant gap activity release into containment and thus considered a loss of the fuel clad barrier. A reading on containment radiation monitors greater than 1.0E+03 R/hr is indicative of significant fuel activity and thus considered a potential loss of Containment (ref. 2).

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. EPIP-2-16 Core Damage Estimation

**Barrier:** Reactor Coolant System

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** E. Isolation Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** E. Isolation Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier |

**Basis:**

Generic

This~~ese~~ threshold~~s~~ addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost~~ or potentially lost~~. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost~~ or potentially lost~~.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

**Ginna Basis Reference(s):**

None

**Barrier:** Reactor Coolant System

**Category:** F. Judgment

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 4. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier |

**Basis:**

Generic

Th~~ese~~is threshold~~s~~ address~~es~~ any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is ~~lost or~~ potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered ~~lost or~~ potentially lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

**Ginna Basis Reference(s):**

None

**Barrier:** Containment

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Containment

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 1. RED path condition exists F-0.5 Containment |

**Basis:**

Generic

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment.

Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this threshold is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.

There is no Loss threshold associated with this item.

Plant-Specific

Critical Safety Function Status Tree (CSFST) Containment-RED path is given in F-0.5 and is entered if Containment pressure is equal to or greater than 60 psig (ref. 1).

This threshold is indicative of a loss of both RCS and Fuel Clad barriers. This combination of conditions would be expected to require the declaration of a General Emergency.

**Ginna Basis Reference(s):**

1. CSFST for F-0.5 Containment

**Barrier:** Containment

**Category:** B. Core Cooling / Heat Removal

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 2. Core Exit TCs **cannot** be restored < 1,200°F within 15 min. |

**Basis:**

Generic

There is no Loss threshold associated with this item.

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

The function restoration procedures are those emergency operating procedures 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.

~~[*For units using the CSF status trees, a direct correlation to those status trees can be made if the effectiveness of the restoration procedures is also evaluated as stated below.*]~~

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

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

~~Potential Loss Threshold B~~

~~[*The reactor vessel level chosen should be consistent with the emergency response guides applicable to the facility.*]~~

Plant-Specific

Core Exit Thermocouples (TCs) reading in excess of 1200°F corresponds to the CSFST Core Cooling RED path entry condition (ref. 1). Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation (ref. 2). Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery. Events that result in Core Exit TC readings above the loss threshold are classified severe accidents and lead to entry into Severe Accident Management Guidelines (ref. 3).

Events that result in Core Exit TC readings above the Fuel Clad loss threshold are classified severe accidents and lead to entry into Severe Accident Management Guidelines and signify possible core overheating to the point that clad ballooning/collapse may occur and portions of the core may have melted (ref. 3).

It must also be assumed the loss of RCS inventory is a result of a loss of the RCS barrier. These conditions, if not mitigated, can lead to core melt which in turn may result in a loss of containment. Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The phrase "cannot be restored <" implies Core Exit TC readings have exceeded the threshold temperature and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful (ref. 4). Whether or not guidance is effective should be apparent within fifteen minutes. The ED should make the declaration as soon as it is determined the guidance has not been or will not be effective in restoring temperature below the threshold.

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. EPIP-2-16 Core Damage Estimation

3. Ginna Severe Accident Management Guidelines
4. FR-C.1 Response to Inadequate Core Cooling

**Barrier:** Containment

**Degradation Threat:** Potential Loss

**Category:** B. Core Cooling / Heat Removal

**Threshold:**

| |
|---|
| 3. Core Exit TCs ≥ 700°F<br>**AND**<br>RVLIS level **cannot** be restored > 52% [> 55% adverse CNMT] with no RCPs running within 15 min. |

**Basis:**

Generic

There is no Loss threshold associated with this item.

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

The function restoration procedures are those emergency operating procedures 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.

~~[For units using the CSF status trees, a direct correlation to those status trees can be made if the effectiveness of the restoration procedures is also evaluated as stated below.]~~

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

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

~~Potential Loss Threshold B~~

~~[The reactor vessel level chosen should be consistent with the emergency response guides applicable to the facility.]~~

Plant-Specific

Core Exit Thermocouples (TCs) reading in excess of 700°F corresponds to the CSFST Core Cooling ORANGE path entry criteria (ref. 1). Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation (ref. 2). RCS superheat, as indicated by Core Exit TCs reading in excess of 700°F, signals the transition from a subcooled to a superheated regime. In a superheated regime, heat transfer mechanics are not as efficient as the subcooled condition and could lead to a rapid rise in clad temperatures. Valid indication of superheat is a potential Fuel Clad barrier loss condition because the possible rapid rise in clad temperatures may lead to clad failure.

This threshold indicates: subcooling has been lost (Core Exit TC readings ≥ 700°F), the core is uncovered and some fuel clad damage may be occurring (ineffective functional restoration procedures) (ref. 1, 3). It must be assumed that the loss of RCS inventory is a result of a loss of the RCS barrier.

These conditions, if not mitigated, can lead to core melt which in turn may result in a loss of containment. Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The phrase "cannot be restored <" implies Core Exit TC readings have exceeded the threshold temperature and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful (ref. 3). Whether or not guidance is effective should be apparent within fifteen minutes. The ED should make the declaration as soon as it is determined the guidance has not been or will not be effective in restoring temperature below the threshold.

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. EPIP-2-16 Core Damage Estimation
3. FR-C.2 Response to Degraded Core Cooling
4. Drawing 03021-0687 Reactor Vessel Level Monitoring System Elevations

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

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

**Basis:**

Generic

~~Loss Thresholds A and B~~

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

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

~~Potential Loss Threshold A~~

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

~~Potential Loss Threshold B~~

~~Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to the RED path in potential loss threshold 1.A above and may be declared by those sites using CSFSTs.~~

~~Potential Loss Threshold C~~

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

Plant-Specific

UFSAR Figure 15.6-34 describes containment pressure response for a large break LOCA (ref. 1). Containment pressure peaks at approximately 45 psig at approximately 25 seconds after event initiation.

**Ginna Basis Reference(s):**

1. UFSAR Figure 15.6-34 Containment Pressure Used for the R.E. Ginna Best-Estimate Large Break LOCA

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

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

**Basis:**

Generic

~~Loss Thresholds A and B~~

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

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

~~Potential Loss Threshold A~~

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

~~Potential Loss Threshold B~~

~~Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to the RED path in potential loss threshold 1.A above and may be declared by those sites using CSFSTs.~~

~~Potential Loss Threshold C~~

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

Plant-Specific

The containment pressure and temperature response and containment sump water temperature response versus time are given in UFSAR Figures 6.2-1 through 6.2-6 for the most severe LOCAs (ref. 1).

**Ginna Basis Reference(s):**

1. UFSAR Figures 6.2-1 through 6.2-6

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 3. Ruptured S/G is also faulted outside of containment |

**Basis:**

Generic

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

Users should realize that th~~e two loss~~is threshold and containment ~~Potential~~ loss C.4~~s~~ could be considered redundant. This was recognized during the development process. The inclusion of a~~n~~ threshold that uses Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

This threshold results in a ~~NO~~UE for smaller breaks that; (1) do not exceed the normal charging capacity threshold in RCS leak rate barrier Potential Loss threshold, or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss threshold. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this threshold would exist in conjunction with RCS barrier thresholds and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

~~Loss Threshold A~~

This threshold addresses the condition in which a ruptured steam generator is also faulted. This condition represents a bypass of the RCS and containment barriers and is a subset of the containment ~~Potential~~ loss C.4~~second threshold~~. In conjunction with RCS leak rate barrier loss threshold, this would always result in the declaration of a Site Area Emergency.

~~Loss Threshold B~~

~~This threshold addresses SG tube leaks that exceed 10 gpm in conjunction with an UNISOLABLE release path to the environment from the affected steam generator. The threshold for establishing the UNISOLABLE secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SG tube rupture with concurrent loss of off-site power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways.~~

~~These pathways do not meet the intent of an UNISOLABLE release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs.~~

~~[*The leakage threshold for this* threshold *has been increased with Revision 3. In the earlier revision, the threshold was leakage greater than T/S allowable. Since the prior revision, many plants have implemented reduced steam generator T/S limits (e.g., 150 gpd) as a defense in depth associated with alternate steam generator plugging criteria. The 150 gpd threshold is deemed too low for use as an emergency threshold. A pressure boundary leakage of 10 gpm was used as the threshold in IC SU5, RCS Leakage, and is deemed appropriate for this* threshold.]~~

Plant-Specific

A faulted S/G means the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized (ref. 1). A ruptured S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection (ref. 2).

Definitions:

**Faulted**

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

**Ruptured**

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

**Ginna Basis Reference(s):**

1. E-2 Faulted Steam Generator Isolation
2. E-3 Steam Generator Tube Rupture

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 4. Primary-to-secondary leakrate > 10 gpm<br>**AND**<br>Unisolable prolonged steam release from affected S/G to the environment |

**Basis:**

Generic

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

Users should realize that the ~~two~~ this loss threshold and containment loss C.3~~s~~ could be considered redundant. This was recognized during the development process. The inclusion of an threshold that uses Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

This threshold results in a ~~NO~~UE for smaller breaks that; (1) do not exceed the normal charging capacity threshold in RCS leak rate barrier Potential Loss threshold, or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss threshold. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this threshold would exist in conjunction with RCS barrier thresholds and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

~~Loss Threshold A~~

~~This threshold addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers and is a subset of the second threshold. In conjunction with RCS leak rate barrier loss threshold, this would always result in the declaration of a Site Area Emergency.~~

~~Loss Threshold B~~

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

~~[The leakage threshold for this~~ threshold ~~has been increased with Revision 3. In the earlier revision, the threshold was leakage greater than T/S allowable. Since the prior revision, many plants have implemented reduced steam generator T/S limits (e.g., 150 gpd) as a defense in depth associated with alternate steam generator plugging criteria. The 150 gpd threshold is deemed too low for use as an emergency threshold. A pressure boundary leakage of 10 gpm was used as the threshold in IC SU5, RCS Leakage, and is deemed appropriate for this~~ threshold.~~]~~

Plant-Specific

Cooldowns conducted to allow controlled isolation of the affected S/G per emergency procedures are not considered prolonged releases. The criterion for prolonged release is met if the objective of E-3 to isolate the affected S/G cannot be met (ref. 2).

An ARV or Safety valve performing as designed is not considered a "failed" barrier.

Definitions:

**Unisolable**

A breach or leak that cannot be promptly isolated from the Main Control Board.

**Ginna Basis Reference(s):**

1. ECA-1.2 LOCA Outside Containment
2. E-3 Steam Generator Tube Rupture
3. F-0.2 Core Cooling

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 4. Containment pressure ≥ 60 psig and rising |

**Basis:**

Generic

~~Loss Thresholds A and B~~

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

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

~~Potential Loss Threshold A~~

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

~~Potential Loss Threshold B~~

~~Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to the RED path in potential loss threshold 1.A above and may be declared by those sites using CSFSTs.~~

~~Potential Loss Threshold C~~

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

Plant-Specific

This threshold is the containment design pressure and is in excess of that expected from the design basis loss of coolant accident (LOCA) (ref. 1, 2). Proper actuation and operation of the containment spray system when required should maintain containment pressure well below the design pressure. The peak containment pressure of 45 psig occurs ~ 25 seconds after event initiation for the most limiting design basis LOCA (ref. 3). The pressure-time responses for the spectrum of LOCAs considered in the plant design basis are described in Section 15 of the UFSAR, Accident Analyses. The threshold is therefore indicative of a loss of both RCS and Fuel Clad barriers in that it should not be reached without severe core degradation (metal-water reaction) or failure to scram in combination with RCS breach. This condition would be expected to require the declaration of a General Emergency.

**Ginna Basis Reference(s):**

1. CSFST for F.0.5 Containment
2. UFSAR 3.1.2.2.7 General Design Criterion 16 – Containment Design
3. UFSAR Figure 15.6-34 Containment Pressure Used for the R.E. Ginna Best-Estimate Large Break LOCA

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 5. Containment hydrogen concentration ≥ 4% |

**Basis:**

Generic

~~Loss Thresholds A and B~~

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

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

~~Potential Loss Threshold A~~

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

~~Potential Loss Threshold B~~

Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to ~~the RED path in~~ containment potential loss threshold A.1~~1.A above and may be declared by those sites using CSFSTs~~.

~~Potential Loss Threshold C~~

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

Plant-Specific

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 Containment. However, Containment monitoring (CH-EPIP-CVH2) and/or sampling should be performed to verify this assumption (ref. 1). A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4.1% (rounded to 4%) by volume (ref. 2).

After a LOCA, the containment atmosphere is a homogeneous mixture of steam, air, solid and gaseous fission products, hydrogen, and water droplets containing boron and sodium hydroxide. During and following a LOCA, the hydrogen concentration in the Containment results from radiolytic decomposition of water, metal-water reaction, and aluminum/zinc reaction with the spray solution (ref. 2). If hydrogen concentration reaches or exceeds the lower flammability limit (4%) in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Clad and RCS barriers with the potential loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

**Ginna Basis Reference(s):**

1. SACRG-1 Severe Accident Control Room Guideline Initial Response
2. UFSAR 1.5.10 Development of Containment Hydrogen Recombiner

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 6. Containment pressure ≥ 28 psig and < two CRFC units and one CS pump operating per design |

**Basis:**

Generic

~~Loss Thresholds A and B~~

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

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

~~Potential Loss Threshold A~~

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

~~Potential Loss Threshold B~~

~~Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to the RED path in potential loss threshold 1.A above and may be declared by those sites using CSFSTs.~~

~~Potential Loss Threshold C~~

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

Plant-Specific

Two means of post accident containment heat removal are provided; Containment Spray System and Containment Recirc Fan Cooler (CRFC) units. At least one train of each of these systems is required to provide sufficient steam-condensing capacity to ensure against containment overstress and to remove residual and chemical heat (ref. 1, 2).

The CRFC system is comprised of four CRFC units, two of which are required in the post accident condition (ref. 3, 4). Each containment aircooling unit consists of cooling coils, accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path. Following an SI actuation signal, CRFC System fans are designed to start automatically (ref. 4).

Each of two containment spray trains consists of a spray pump, spray header, nozzles, valves, piping, instruments, and controls to ensure an operable flow path capable of taking suction from the RWST upon an actuation signal (ref. 4).

During a steam line break or LOCA, a minimum of two CRFC units and one Containment Spray (CS) pump are required to maintain peak pressure and temperature below design limits (ref. 4).

The containment hi-hi pressure setpoint (28 psig) is the pressure at which the equipment should actuate and begin performing its function (ref. 5).

**Ginna Basis Reference(s):**

1. UFSAR Section 6.2.2 Containment Heat Removal Systems
2. UFSAR Section 6.2.1.2.3 Secondary System Pipe Break Analysis
3. UFSAR Section 6.2.2.1.3 Design Evaluation
4. Technical Specifications B 3.6. Containment Systems
5. CSFST for F-0.5 Containment

**Barrier:** Containment

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Containment

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 7. Containment radiation monitor R-29/R-30 reading > 1.0E+03 R/hr |

**Basis:**

Generic

There is no Loss threshold associated with this item.

The site specific reading is a value which indicates significant fuel damage well in excess of the thresholds associated with both loss of Fuel Clad and loss of RCS barriers. ~~As stated in Section 3.8, a~~ major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of fuel clad allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted.

~~[*NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. Unless there is a (site specific) analysis justifying a higher value, it is recommended that a radiation monitor reading corresponding to 20% fuel clad damage be specified here.*~~]

Plant-Specific

Containment radiation is indicated on R-29 and R-30 (ref. 1).

R-29 & R-30 alert alarms at 10 R/hr, indicative of a significant RCS breach (LOCA) in containment (~0.1% gap activity). The R-29 & R-30 high alarm setpoint is set at 100 R/hr and is indicative of a significant gap activity release into containment and thus considered a loss of the fuel clad barrier. A reading on containment radiation monitors greater than 1.0E+03 R/hr is indicative of significant fuel activity and thus considered a potential loss of Containment (ref. 2).

The containment radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier.

NUREG-1228 "Source Term Estimations During Incident Response to Severe Nuclear Power Plant Accidents" states that such readings do not exist when the amount of clad damage is less than 20% (ref. 3). A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure into the reactor coolant has occurred. Regardless of whether the Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Containment barrier.

The reading is higher than that specified for Fuel Clad barrier Loss #3 and RCS barrier Loss #3. Containment radiation readings at or above the Containment barrier potential loss threshold, therefore, signify a loss of two fission product barriers and potential loss of a third, indicating the need to upgrade the emergency classification to a General Emergency.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. EPIP-2-16 Core Damage Assessment Estimation
3. NUREG-1228 Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents

**Barrier:** Containment

**Category:** E. Isolation Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 5. Failure of **all** valves in **ANY one** line to close<br>**AND**<br>Direct downstream pathway to the environment exists after containment isolation signal |

**Basis:**

Generic

This threshold addresses incomplete containment isolation that allows direct release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no Potential Loss threshold associated with this item.

Plant-Specific

None

**Ginna Basis Reference(s):**

1. EOP Attachment 27 Attachment Automatic Action Verification
2. EOP Attachment 3 Attachment CI/CVI

**Barrier:** Containment

**Category:** E. Isolation Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Containment

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier |

**Basis:**

Generic

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

The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

**Ginna Basis Reference(s):**

None

**Barrier:** Containment

**Category:** F. Judgment

**Degradation Threat:** Potential Loss

**Threshold:**

8. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

**Basis:**

Generic

Th<u>is</u>~~ese~~ threshold~~s~~ address<u>es</u> any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is ~~lost or~~ potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered ~~lost or~~ potentially lost.

The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

**Ginna Basis Reference(s):**

None

# ATTACHMENT (2)

# EAL TECHNICAL BASES

GINNA STATION

CONTROLLED COPY NUMBER ________

PROCEDURE NO. EPAD X-X REV. No. [Draft H]

# EMERGENCY ACTION LEVEL TECHNICAL BASES DOCUMENT

Coordinator, Emergency Preparedness

RESPONSIBLE MANAGER

xx/xx/xx

EFFECTIVE DATE

[Draft H 12/20/11]

CATEGORY x.x

THIS PROCEDURE CONTAINS 278 PAGES

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## ACRONYMS & ABBREVIATIONS

AC .......... Alternating Current
APRM .......... Average Power Range Meter
ATWS .......... Anticipated Transient Without Scram
CCW .......... Component Cooling Water
CDE .......... Committed Dose Equivalent
CE .......... Combustion Engineering
CFR .......... Code of Federal Regulations
CNMT .......... Containment
CSF .......... Critical Safety Function
CSFST .......... Critical Safety Function Status Tree
DC .......... Direct Current
DHR .......... Decay Heat Removal
Disch .......... Discharge
DOT .......... Department of Transportation
EAL .......... Emergency Action Level
ECCS .......... Emergency Core Cooling System
ECL .......... Emergency Classification Level
ED .......... Emergency Director
EOF .......... Emergency Operations Facility
EOP .......... Emergency Operating Procedure
EPA .......... Environmental Protection Agency
EPG .......... Emergency Procedure Guideline
EPRI .......... Electric Power Research Institute
ERG .......... Emergency Response Guideline
EPIP .......... Emergency Plan Implementing Procedure
ESF .......... Engineered Safety Feature
ESW .......... Emergency Service Water
FAA .......... Federal Aviation Administration
FBI .......... Federal Bureau of Investigation
FEMA .......... Federal Emergency Management Agency
FSAR .......... Final Safety Analysis Report
GE .......... General Emergency
HOO .......... Headquarters (NRC) Operations Officer
HPSI .......... High Pressure Safety Injection
IC .......... Initiating Condition
IPEEE .......... Individual Plant Examination of External Events (Generic Letter 88-20)

## ACRONYMS & ABBREVIATIONS (continued)

ISFSI ........ Independent Spent Fuel Storage Installation
Keff ........ Effective Neutron Multiplication Factor
LCO ........ Limiting Condition of Operation
LER ........ Licensee Event Report
LOCA ........ Loss of Coolant Accident
LPSI ........ Low Pressure Safety Injection
LWR ........ Light Water Reactor
MSIV ........ Main Steam Isolation Valve
MSL ........ Main Steam Line
mR ........ milliRoentgen
MW ........ Megawatt
MWS ........ Miscellaneous Waste 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
NUMARC ........ Nuclear Management and Resources Council
OBE ........ Operating Basis Earthquake
OCA ........ Owner Controlled Area
ODCM ........ Off-site Dose Calculation Manual
ORO ........ Off-site Response Organization
OTCC ........ Once Through Core Cooling
PA ........ Protected Area
PAG ........ Protective Action Guideline
POAH ........ Point of Adding Heat
PRA/PSA ........ Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR ........ Pressurized Water Reactor
PSIG ........ Pounds per Square Inch Gauge
R ........ Roentgen
RCC ........ Reactor Control Console
RCIC ........ Reactor Core Isolation Cooling
RCS ........ Reactor Coolant System
rem ........ Roentgen Equivalent Man
RETS ........ Radiological Effluent Technical Specifications
RPS ........ Reactor Protection System
RPV ........ Reactor Pressure Vessel

## ACRONYMS & ABBREVIATIONS (continued)

RVLIS ........ Reactor Vessel Level Indicating System
RWCU ........ Reactor Water Cleanup
SAE ........ Site Area Emergency
SBO ........ Station Blackout
SG ........ Steam Generator
SI ........ Safety Injection
SPDS ........ Safety Parameter Display System
SRO ........ Senior Reactor Operator
SSE ........ Safe Shutdown Earthquake
TEDE ........ Total Effective Dose Equivalent
TOAF ........ Top of Active Fuel
TSC ........ Technical Support Center
UE ........ Unusual Event
WE ........ Westinghouse Electric
WOG ........ Westinghouse Owners Group
WRNGM ........ Wide Range Noble Gas Monitor

## 1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for the R. E. Ginna Nuclear Power Plant (Ginna). It should be used to facilitate review of the Ginna EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EPIP-1-0 "Ginna Station Event Evaluation and Classification" and the Emergency Action Level Matrix may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training, for explaining event classifications to off-site officials, and for facilitating regulatory review and approval of the classification scheme.

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 in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

## 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 Ginna 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) Revision 4 was 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 5 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL FAQs. Using NEI 99-01 Revision 5 Final, February 2008 (ADAMS Accession Number ML080450149), Ginna conducted an EAL implementation upgrade project that produced the EALs discussed herein.

## 2.2 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon 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. "Loss" means the barrier no longer assures containment of radioactive materials; "Potential Loss" 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 zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
C. Containment (CNMT): 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.

## 2.3 Emergency Classification Based on Fission Product Barrier Degradation

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

*Unusual Event:*

*Any loss or any potential loss of Containment*

*Alert:*

*Any loss or any potential loss of either Fuel Clad or RCS*

*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*

## 2.4 EAL Relationship to EOPs and Critical Safety Function Status

Where possible, the EALs have been made consistent with and utilize the conditions defined in the Ginna Emergency Operating Procedure (EOP) network. While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events which indicate reactor plant safety and/or fission product barrier integrity are threatened. When these symptoms are clearly representative of one of the NEI 99-01 Rev. 5 Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the EOPs, classification of emergencies using these EALs is not dependent upon EOP entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

## 2.5 Symptom-Based vs. Event-Based Approach

To the extent possible, the EALs are symptom-based. That is, the action level threshold is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. However, a purely symptom-based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

## 2.6 EAL Organization

The Ginna EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - o EALs applicable under <u>all</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - o EALs applicable only under <u>hot</u> 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.
  - o EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refuel 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 of the above three groups, assignment of EALs to categories/subcategories – category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The Ginna EAL categories/subcategories and their relationship to NEI 99-01 Rev. 5 Recognition Categories are listed below.

## EAL Groups, Categories and Subcategories

| EAL Group/Category | EAL Subcategory |
|---|---|
| Any Operating Mode: | |
| R – Abnormal **R**ad Release / Rad Effluent | 1 – Offsite Rad Conditions<br>2 – Onsite Rad Conditions & Spent Fuel Events<br>3 – CR/CAS/SAS Rad |
| H – Hazards and Other Conditions Affecting Plant Safety | 1 – Natural or Destructive Phenomena<br>2 – Fire or Explosion<br>3 – Hazardous Gas<br>4 – Security<br>5 – Control Room Evacuation<br>6 – Judgment |
| E – ISFSI | None |
| Hot Conditions: | |
| S – **S**ystem Malfunction | 1 – Loss of AC Power<br>2 – Loss of DC Power<br>3 – Criticality & RPS Failure<br>4 – Inability to Reach or Maintain Shutdown Conditions<br>5 – Instrumentation<br>6 – Communications<br>7 – Fuel Clad Degradation<br>8 – RCS Leakage |
| F – **F**ission Product Barrier Degradation | None |
| Cold Conditions: | |
| C – **C**old Shutdown / Refueling System Malfunction | 1 – Loss of AC Power<br>2 – Loss of DC Power<br>3 – RCS Level<br>4 – RCS Temperature<br>5 – Communications<br>6 – Inadvertent Criticality |

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 Sections 2.7 & 2.8, and Attachments 1 & 2 of this document for such information.

## 2.7 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, E, C, H, S and F) 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:

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

<u>Initiating Condition (IC)</u>

Site-specific description of the generic IC given in NEI 99-01 Rev. 5.

<u>EAL Identifier (enclosed in rectangle)</u>

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, E, C, H, S or F)
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)

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 Shutdown, 4 - Hot Standby, 5 - Cold Shutdown, 6 - Refuel, D - Defueled, or All. (See Section 2.8 for operating mode definitions)

Basis:

A Generic basis section provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 5. This is followed by a Plant-Specific basis section that provides Ginna-relevant information concerning the EAL. If the EAL wording contains a defined term, the definition of the term is included at the end of the plant-specific basis discussion.

Ginna Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.8 Operating Mode Applicability (Based on Technical Specifications Table 1.1-1)

1 Power Operation

Reactor shutdown margin is less than Technical Specification minimum required ($K_{eff} \geq 0.99$) and greater than 5% rated thermal power (excluding decay heat).

2 Startup

Reactor shutdown margin is less than Technical Specification minimum required ($K_{eff} \geq 0.99$) and less than or equal to 5% rated thermal power (excluding decay heat).

3 Hot Shutdown

Reactor shutdown margin greater than Technical Specification minimum required ($K_{eff} < 0.99$) with coolant temperature (Tavg) greater than or equal to 350°F.

4 Hot Standby

Reactor shutdown margin greater than Technical Specification minimum required ($k_{eff} < 0.99$) with coolant temperature (Tavg) less than 350°F and greater than 200°F (all reactor vessel head closure bolts fully tensioned).

5 Cold Shutdown

Reactor shutdown margin greater than Technical Specification minimum required ($K_{eff} < 0.99$) with coolant temperature (Tavg) less than or equal to 200°F (all reactor vessel head closure bolts fully tensioned).

6 Refuel

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

D Defueled

All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).

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.

2.9 Validation of Indications, Reports and Conditions

All EALs and Fission Product Barrier thresholds assume valid indications. All emergency classifications shall be based upon valid indications, reports or conditions. 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.

## 2.10 Planned vs. Unplanned Events

Planned evolutions involve preplanning to address the limitations imposed by the condition, the performance of required surveillance testing, and the implementation of specific controls prior to knowingly entering the condition in accordance with the specific requirements of the site's Technical Specifications. Activities, planned or unplanned, which cause the site to operate beyond what is allowed by the site's Technical Specifications may result in an EAL threshold being met or exceeded. Planned evolutions to test, manipulate, repair or perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

## 2.11 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined in other situations, further analyses may be necessary (e.g., coolant radiochemistry following an ATWS event, plant structural examination following an earthquake, etc.). Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

Existing guidance for classifying transient events addresses the period of time of event recognition and classification (15 minutes). However, in cases when EAL declaration criteria may be met momentarily during the normal expected response of the plant, declaration requirements should not be considered to be met when the conditions are a part of the designed plant response, or result from appropriate Operator actions.

There may be cases in which a plant condition that exceeded an EAL was not recognized at the time of occurrence but is identified well after the condition has occurred (e.g., as a result of routine log or record review), and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, should be applied.

### 2.12 Multiple Simultaneous Events and Imminent EAL Thresholds

When multiple simultaneous events occur, the emergency classification level is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency. Further guidance is provided in RIS 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.

Although the majority of the EALs provide very specific thresholds, the Emergency Director (ED) must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the ED, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded. While this is particularly prudent at the higher emergency classes (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

### 2.13 Emergency Classification Level Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing. A combination approach involving recovery from General Emergencies and some Site Area Emergencies and termination from Unusual Events, Alerts, and certain Site Area Emergencies causing no long term plant damage appears to be the best choice. Downgrading to lower emergency classification levels adds notifications but may have merit under certain circumstances.

## 3.0 REFERENCES

### 3.1 Developmental

3.1.1 NEI 99-01 Rev. 5 Final, Methodology for Development of Emergency Action Levels, February 2008, ADAMS Accession Number ML080450149

3.1.2 NRC Regulatory Issue Summary (RIS) 2003-18, Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels Revision 4, Dated January 2003 (December 12, 2005)

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

### 3.2 Implementing

3.2.1 EPIP-1-0, Ginna Station Event Evaluation and Classification

3.2.2 EAL Comparison Matrix

3.2.3 EAL Matrix

### 3.3 Commitments

None

## 4.0 DEFINITIONS (ref. 3.1.1 except as noted)

**Affecting Safe Shutdown**

Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable hot or cold shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

> Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in hot shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is not "affecting safe shutdown."
>
> Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in cold shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is "affecting safe shutdown."

**Airliner/Large Aircraft**

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

**Bomb**

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

**Civil Disturbance**

A group of people violently protesting station operations or activities at the site.

**Confinement Boundary**

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

**Containment Closure**

The site-specific procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to Ginna, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in O-2.3.1A, provide a functional barrier to fission product release.

**Explosion**

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

**Extortion**

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

**Faulted**

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

**Fire**

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

**Hostage**

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

**Hostile Action**

An act toward Ginna 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 Ginna. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

**Hostile Force**

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

**Imminent**

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

**Intrusion**

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.

**Normal Levels**

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

**Normal Plant Operations**

Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

**Owner Controlled Area**

The site-specific facilities and property outside the the security Protected Area fence.

**Projectile**

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

**Protected Area**

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

**RCS Intact**

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

**Ruptured**

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

**Sabotage**

Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of sabotage until this determination is made by security supervision.

**Safety-Related Structures, Systems and Components** (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 Boundary**

The Site Boundary is approximately a 0.3-mile radius around the reactor.

**Strike Action**

Work stoppage within the Protected Area by a body of workers to enforce compliance with demands made on Ginna. The strike action must threaten to interrupt Normal Plant Operations.

**Unisolable**

A breach or leak that cannot be promptly isolated from the Main Control Board.

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Valid**

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

**Visible Damage**

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

**Vital Area**

Any area, normally within the Ginna Protected Area, that contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

## 5.0 GINNA-TO-NEI 99-01 EAL CROSS-REFERENCE

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

| GINNA | NEI 99-01 | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| RU1.1 | AU1 | 1, 2 |
| RU1.2 | AU1 | 3 |
| RU2.1 | AU2 | 1 |
| RU2.2 | AU2 | 2 |
| RA1.1 | AA1 | 1 |
| RA1.2 | AA1 | 3 |
| RA2.1 | AA2 | 2 |
| RA2.2 | AA2 | 1 |
| RA3.1 | AA3 | 1 |
| RS1.1 | AS1 | 1 |
| RS1.2 | AS1 | 2 |
| RS1.3 | AS1 | 4 |
| RG1.1 | AG1 | 1 |
| RG1.2 | AG1 | 2 |
| RG1.3 | AG1 | 4 |
| EU1.1 | E-HU1 | 1 |
| CU1.1 | CU3 | 1 |

| GINNA | NEI 99-01 | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| CU2.1 | CU7 | 1 |
| CU3.1 | CU1 | 1 |
| CU3.2 | CU2 | 1 |
| CU3.3 | CU2 | 2 |
| CU4.1 | CU4 | 1 |
| CU4.2 | CU4 | 2 |
| CU5.1 | CU6 | 1, 2 |
| CU6.1 | CU8 | 2 |
| CA1.1 | CA3 | 1 |
| CA3.1 | CA1 | 1, 2 |
| CA4.1 | CA4 | 1, 2 |
| CS3.1 | CS1 | 1 |
| CS3.2 | CS1 | 2 |
| CS3.3 | CS1 | 3 |
| CG3.1 | CG1 | 1 |
| FU1.1 | FU1 | 1 |
| FA1.1 | FA1 | 1 |
| FS1.1 | FS1 | 1 |
| FG1.1 | FG1 | 1 |
| HU1.1 | HU1 | 1 |
| HU1.2 | HU1 | 2 |
| HU1.3 | HU1 | 3 |
| HU1.4 | HU1 | 4 |

| GINNA | NEI 99-01 | |
|---|---|---|
| EAL | IC | Example EAL |
| HU1.5 | HU1 | 5 |
| HU2.1 | HU2 | 1 |
| HU2.2 | HU2 | 2 |
| HU3.1 | HU3 | 1 |
| HU3.2 | HU3 | 2 |
| HU4.1 | HU4 | 1, 2, 3 |
| HU6.1 | HU5 | 1 |
| HA1.1 | HA1 | 1 |
| HA1.2 | HA1 | 2 |
| HA1.3 | HA1 | 3 |
| HA1.4 | HA1 | 4 |
| HA1.5 | HA1 | 6 |
| HA1.6 | HA1 | 5 |
| HA2.1 | HA2 | 1 |
| HA3.1 | HA3 | 1 |
| HA4.1 | HA4 | 1, 2 |
| HA5.1 | HA5 | 1 |
| HA6.1 | HA6 | 1 |
| HS4.1 | HS4 | 1 |
| HS5.1 | HS2 | 1 |
| HS6.1 | HS3 | 1 |
| HG4.1 | HG1 | 1 |
| HG4.2 | HG1 | 2 |

| GINNA | NEI 99-01 | |
|---|---|---|
| EAL | IC | Example EAL |
| HG6.1 | HG2 | 1 |
| SU1.1 | SU1 | 1 |
| SU3.1 | SU8 | 2 |
| SU4.1 | SU2 | 1 |
| SU5.1 | SU3 | 1 |
| SU6.1 | SU6 | 1, 2 |
| SU7.1 | SU4 | 2 |
| SU7.2 | SU4 | 1 |
| SU8.1 | SU5 | 1, 2 |
| SA1.1 | SA5 | 1 |
| SA3.1 | SA2 | 1 |
| SA5.1 | SA4 | 1 |
| SS1.1 | SS1 | 1 |
| SS2.1 | SS3 | 1 |
| SS3.1 | SS2 | 1 |
| SS5.1 | SS6 | 1 |
| SG1.1 | SG1 | 1 |
| SG3.1 | SG2 | 1 |

## 6.0 ATTACHMENTS

6.1 Attachment 1, Emergency Action Level Technical Bases

6.2 Attachment 2, Fission Product Barrier Loss / Potential Loss Matrix and Basis

# ATTACHMENT 1

# EMERGENCY ACTION LEVEL
# TECHNICAL 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. Offsite Rad Conditions

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. Onsite Rad Conditions & Spent Fuel Events

Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials or those levels which may preclude access to vital plant areas also warrant emergency classification.

3. CR/CAS Rad

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

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **ANY** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer

**EAL:**

**RU1.1 Unusual Event**

**ANY** gaseous or liquid monitor reading > Table R-1 column "UE" for ≥ 60 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| **Gaseous** | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 μC/cc | 1.8E+1 μC/cc | 1.8E+0 μC/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 μC/cc | 2.1E+0 μC/cc | 2.1E-1 μC/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 μC/cc | 5.7E+1 μC/cc | 5.7E+0 μC/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| **Liquid** | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm with no isolation |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | 9.2E+4 cpm with no isolation |

**Mode Applicability:**

All

**Basis:**

Generic

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 will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

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

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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

This EAL is also intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

Plant-Specific

Monitor indications are derived from release limits determined by the ODCM methodology and specified offsite dose criteria (ref. 1). These values are summarized in Reference 2.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

**Ginna Basis Reference(s):**

1. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
2. CALC-2011-0020, NEI 99-01 Technical Basis for the Ginna Effluent Monitor Emergency Action Levels (EALs)
3. NEI 99-01 AU1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **ANY** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer

**EAL:**

| | |
|---|---|
| **RU1.2** | **Unusual Event** |
| Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x P-9 limits for ≥ 60 min. (Note 2) | |

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Mode Applicability:**

All

**Basis:**

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2x P-9 (ODCM) limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x P-9 (ODCM) for 30 minutes does not meet the threshold.

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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

Plant-Specific

Offsite Dose Calculation Manual (ODCM) release limits are specified in Technical Procedure P-9 (ref. 1).

Releases in excess of two times the site ODCM (ref. 2) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the ODCM limit for 30 minutes does not exceed this initiating condition. Further, the ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
3. NEI 99-01 AU1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **ANY** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer

**EAL:**

**RA1.1 Alert**

**ANY** gaseous monitor reading > Table R-1 column "Alert" for ≥ 15 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| **Gaseous** | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 µC/cc | 1.8E+1 µC/cc | 1.8E+0 µC/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 µC/cc | 2.1E+0 µC/cc | 2.1E-1 µC/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 µC/cc | 5.7E+1 µC/cc | 5.7E+0 µC/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| **Liquid** | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm with no isolation |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | 9.2E+4 cpm with no isolation |

**Mode Applicability:**

All

**Basis:**

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The value of 1% (10 mrem) of the EPA PAG threshold (in lieu of 200 times the ODCM release rate limit) is specified to provide a realistic escalation path between the Unusual Event and Site Area Emergency classifications for gaseous releases. While these thresholds obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant. Releases should not be prorated or averaged. For example, a release exceeding 600x ODCM for 5 minutes does not meet the threshold.

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

Plant-Specific

The values shown correspond to a dose of 10 mrem in one hour at the site boundary. Monitor indications are calculated (ref. 2) based on annual average X/Q dispersion factors from the ODCM (ref. 1) and a source term representative of accident conditions. For the main steam line monitors (R-31/32), the variability of results based upon the number of ARVs and/or Main Steam Safety Valves precludes the use of any single default value for these monitors. For these cases, adjustments are made for expected flow rates.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

**Ginna Basis Reference(s):**

1. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
2. CALC-2011-0020, NEI 99-01 Technical Basis for the Ginna Effluent Monitor Emergency Action Levels (EALs)
3. NEI 99-01 AA1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **ANY** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer

**EAL:**

**RA1.2 Alert**

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x P-9 limits for ≥ 15 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Mode Applicability:**

All

**Basis:**

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200x ODCM limit are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

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

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage.

Plant-Specific

Offsite Dose Calculation Manual (ODCM) release limits are specified in Technical Procedure P-9 (ref. 1).

Releases in excess of two hundred times the site ODCM (ref. 2) instantaneous limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential significant degradation in the level of safety. The final integrated dose (which is very low in the Alert emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 15 minutes. Therefore, it is not intended that the release be averaged over 15 minutes. For example, a release of 400 times the ODCM limit for 7.5 minutes does not exceed this initiating condition. Further, the ED should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
3. NEI 99-01 AA1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

**RS1.1 Site Area Emergency**

**ANY** gaseous monitor reading > Table R-1 column "SAE" for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RS1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| **Gaseous** | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 μC/cc | 1.8E+1 μC/cc | 1.8E+0 μC/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 μC/cc | 2.1E+0 μC/cc | 2.1E-1 μC/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 μC/cc | 5.7E+1 μC/cc | 5.7E+0 μC/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| **Liquid** | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm<br>with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm<br>with no isolation |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | 9.2E+4 cpm<br>with no isolation |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

The site specific monitor list in Table R-1 includes effluent monitors on all potential release pathways.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The values shown correspond to a dose of 100 mrem in one hour at the site boundary. Monitor indications are calculated (ref. 2) based on annual average X/Q dispersion factors from the ODCM (ref. 1) and a source term representative of accident conditions. For the main steam line monitors (R-31/32), the variability of results based upon the number of ARVs and/or Main Steam Safety Valves precludes the use of any single default value for these monitors. For these cases, adjustments are made for expected flow rates.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

**Ginna Basis Reference(s):**

1. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
2. CALC-2011-0020, NEI 99-01 Technical Basis for the Ginna Effluent Monitor Emergency Action Levels (EALs)
3. NEI 99-01 AS1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**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 | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The 100 mRem TEDE dose is set at 10% of the EPA PAG, while the 500 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Dose assessment may be performed by either manual or computer based methods (ref. 1, 2, 3).

Definitions:

**Site Boundary**

The site boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. EPIP-2-18 Control Room Dose Assessment
2. EPIP-2-5 Emergency Dose Projections - Personal Computer Method
3. EPIP-2-4 Emergency Dose Projections - Manual Method
4. NEI 99-01 AS1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

**RS1.3 Site Area Emergency**

Field survey results indicate closed window dose rates > 100 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary

**OR**

Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the site boundary

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

Real time field surveys and sample analysis is performed by offsite field monitoring teams per EPIP-2-12 "Offsite Surveys" (ref. 1) and assessed for radiological dose consequences per EPIP-2-5 " Emergency Dose Projections - Personal Computer Method" (ref. 2).

Definitions:

**Site Boundary**

The site boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. EPIP-2-12 Offsite Surveys
2. EPIP-2-5 Emergency Dose Projections - Personal Computer Method
3. NEI 99-01 AS1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

**RG1.1 General Emergency**

**ANY** gaseous monitor reading > Table R-1 column "GE" for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RG1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| __Gaseous__ | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 μC/cc | 1.8E+1 μC/cc | 1.8E+0 μC/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 μC/cc | 2.1E+0 μC/cc | 2.1E-1 μC/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 μC/cc | 5.7E+1 μC/cc | 5.7E+0 μC/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| __Liquid__ | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm with no isolation |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | 9.2E+4 cpm with no isolation |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

The monitor list in Table R-1 includes effluent monitors on all potential release pathways.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The values shown correspond to a dose of 1000 mrem in one hour at the site boundary. Monitor indications are calculated (ref. 2) based on annual average X/Q dispersion factors from the ODCM (ref. 1) and a source term representative of accident conditions. For the main steam line monitors (R-31/32), the variability of results based upon the number of ARVs and/or Main Steam Safety Valves precludes the use of any single default value for these monitors. For these cases, adjustments are made for expected flow rates.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

**Ginna Basis Reference(s):**

1. R. E. Ginna Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM)
2. CALC-2011-0020, NEI 99-01 Technical Basis for the Ginna Effluent Monitor Emergency Action Levels (EALs)
3. NEI 99-01 AG1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**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 | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The 1000 mRem TEDE dose is set at 100% of the EPA PAG, while the 5000 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Dose assessment may be performed by either manual or computer based methods (ref. 1, 2, 3).

Definitions:

**Site Boundary**

The site boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. EPIP-2-18 Control Room Dose Assessment
2. EPIP-2-5 Emergency Dose Projections - Personal Computer Method
3. EPIP-2-4 Emergency Dose Projections - Manual Method
4. NEI 99-01 AG1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

**RG1.3 General Emergency**

Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary

**OR**

Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the site boundary

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

Real time field surveys and sample analysis are performed by offsite field monitoring teams per EPIP-2-12 "Offsite Surveys" (ref. 1) and assessed for radiological dose consequences per EPIP-2-5 " Emergency Dose Projections - Personal Computer Method" (ref. 2).

Definitions:

**Site Boundary**

The site boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. EPIP-2-12 Offsite Surveys
2. EPIP-2-5 Emergency Dose Projections - Personal Computer Method
3. NEI 99-01 AG1

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Events

**Initiating Condition:** Unplanned rise in plant radiation levels

**EAL:**

**RU2.1 Unusual Event**

Unplanned water level drop in a reactor refueling pathway as indicated by inability to restore and maintain level > SFP low water level alarm setpoint (Note 3)

**AND**

Area radiation monitor reading rise on **EITHER**:

R-2 Containment

**OR**

R-5 Spent Fuel Pool

Note 3: If loss of water level in the refueling pathway occurs while in Mode 5, 6 or D, consider classification under EALs CU3.1, CU3.2 or CU3.3

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in unplanned increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

The refueling pathway is a combination of cavities, tubes, canals and pools. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

For refueling events where the water level drops below the Reactor Vessel flange classification would be via EAL CU3.1, CU3.2 or CU3.3. This event escalates to an Alert per EAL RA2.1 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-4.

Plant-Specific

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

The SFP is equipped with a level switch (LC-661) that actuates a low level alarm at 20 in. from the top of the SFP (ref. 1). The minimum level per Technical Specifications is 23 feet above the fuel seated in the SFP (ref. 2).

The definition of "... cannot be restored and maintained above..." allows the operator to visually observe the low water level condition, if possible, and to attempt water level restoration instructions as long as water level remains above the top of irradiated fuel.

When the fuel transfer canal is directly connected to the Spent Fuel Pool and refueling cavity, there could exist the possibility of uncovering irradiated fuel in the fuel transfer canal. Therefore, this EAL is applicable to conditions in which irradiated fuel is being transferred to and from the reactor vessel and SFP.

Technical Specifications requires that refueling cavity water level be maintained 23 ft above irradiated fuel seated in the reactor vessel when moving fuel (ref. 3).

Area radiation monitors R-2 and R-5 are located in the proximity of where spent fuel may be located and have been selected to be indicative of a decrease in radiation shielding due to decreasing refueling pathway water level (ref. 4). While a radiation monitor could detect a rise in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not the fuel is uncovered. For example, the reading on an area radiation monitor located on the refueling bridge may rise due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

This event escalates to an Alert if irradiated fuel outside the reactor vessel is uncovered.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. AR-K-29 SFP HI TEMP 115 ºF HI-LO LEVEL 20" 12"
2. Technical Specifications Section 3.7.11 Spent Fuel Pool (SFP) Water Level
3. Technical Specifications Section 3.9.6 Refueling Cavity Water Level
4. P-9 Radiation Monitoring System
5. NEI 99-01 AU2

**Category:** R – Radioactivity Release / Area Radiation

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Events

**Initiating Condition:** Unplanned rise in plant radiation levels

**EAL:**

| | |
|---|---|
| **RU2.2** | **Unusual Event** |
| Unplanned area radiation reading increases by a factor of 1,000 over normal levels | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in unplanned increases in radiation dose rates within plant buildings.

This EAL addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the threshold. The intent is to identify loss of control of radioactive material in any monitored area.

Plant-Specific

Assessment of this EAL may be made with survey readings using portable instruments as well as installed radiation monitors.

Definitions:

**Normal Levels**

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

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. NEI 99-01 AU2

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Events

**Initiating Condition:** Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

**EAL:**

**RA2.1 Alert**

Alarm on **ANY** of the following radiation monitors due to damage to irradiated fuel or loss of water level:

- R-12 Containment Vent Noble Gas
- R-14 Plant Vent Noble Gas
- R-2 Containment
- R-5 Spent Fuel Pool

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.1, RG1.2 or RG1.3.

Plant-Specific

This EAL is defined by the specific areas where irradiated fuel is located such as the refueling cavity, reactor vessel, or spent fuel pool.

The bases for the area radiation alarms include a spent fuel handling accident and are, therefore, appropriate for this EAL. Elevated readings on ventilation monitors may also be indication of a radioactivity release from the fuel, confirming that damage has occurred (ref. 1). However, elevated background at the monitor due to water level lowering may mask elevated ventilation exhaust airborne activity and needs to be considered. However, while radiation monitors may detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Interpretation of these EAL thresholds requires some understanding of the actual radiological conditions present in the vicinity of the monitors.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. NEI 99-01 AA2

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Events

**Initiating Condition:** Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

**EAL:**

| | |
|---|---|
| **RA2.2** | **Alert** |
| A water level drop in a reactor refueling pathway that will result in irradiated fuel becoming uncovered | |

**Mode Applicability:**

All

**Basis:**

Generic

This event represents a loss of control over radioactive material and represents an actual or substantial potential degradation in the level of safety of the plant.

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.1, RG1.2 or RG1.3.

Plant-Specific

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

There is no indirect indication that water level in the spent fuel pool or refueling cavity has dropped to the level of the fuel other than visual observation. Since there is no level indicating system in the fuel transfer canal, visual observation of loss of water level would also be required. If available, video cameras may allow remote observation. Depending on available level indication, the declared threshold may need to be based on indications of makeup rate or lowering in Reactor Coolant Drain Tank (RCDT) level (ref. 1).

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the reactor vessel flange and the top of spent fuel in the SFP. During refueling activities, this maintains sufficient water level in the refueling cavity, fuel transfer canal and SFP. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (ref. 2, 3).

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment.

**Ginna Basis Reference(s):**

1. ER-SFP.1 Loss of Spent Fuel Pool Cooling
2. Technical Specifications Section 3.9.6 Refueling Cavity Water Level
3. Technical Specifications Section 3.7.11 Spent Fuel Pool (SFP) Water Level
4. NEI 99-01 AA2

**Category:** R – Abnormal Rad Release / Rad Effluent

**Subcategory:** 3 – CR/CAS Rad

**Initiating Condition:** Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

**EAL:**

**RA3.1 Alert**

Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:

Control Room (R-1)

**OR**

CAS

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses increased radiation levels that: impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

The cause and/or magnitude of the increase in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved.

Areas requiring continuous occupancy include the Control Room and any other control stations that are staffed continuously, such as the security alarm stations CAS and SAS.

Plant-Specific

The Control Room and Central Alarm Station (CAS) must be continuously occupied in all plant operating modes at Ginna.

Area radiation monitor (ARM) R-1 detects radiation levels in the vicinity of the main Control Room. This ARM alarms at 2 mR/hr giving personnel sufficient warning of changing levels (ref. 1). There is no area radiation monitoring system at Ginna for the CAS. Abnormal radiation levels may be initially detected by routine radiological surveys.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. NEI 99-01 AA3

## Category E – Independent Spent Fuel Storage Installation (ISFSI)

EAL Group: Not Applicable (the EAL in this category is applicable independent of plant operating mode)

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. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

A Notification of 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. This includes classification based on a loaded fuel storage cask/canister confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

A hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HA4.1.

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

**Category:** E – ISFSI

**Subcategory:** Not Applicable

**Initiating Condition:** Damage to a loaded cask confinement boundary

**EAL:**

| | |
|---|---|
| **EU1.1** | **Unusual Event** |
| Damage to a loaded cask confinement boundary | |

**Mode Applicability:**

Not applicable

**Basis:**

Generic

An UE in this EAL is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

Plant-Specific

The Ginna ISFSI utilizes the NUHOMS dry spent fuel storage system.

This EAL addresses any condition which indicates a loss of a cask confinement boundary and thus a potential degradation in the level of safety of the ISFSI. The cask confinement boundary is considered the Dry Shielded Canister (DSC).

Definitions:

**Confinement Boundary**

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

**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.

**Ginna Basis Reference(s):**

1. R. E. Ginna ISFSI USAR
2. NEI 99-01 E-HU1

## Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (RCS 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 RCS integrity, containment closure, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refuel, D – Defueled).

The events of this category pertain to the following subcategories:

1. Loss of AC 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 onsite and offsite power sources for the 480V safeguard buses.

2. Loss of 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 the 125 VDC buses.

3. RCS Level

Reactor Vessel or RCS water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity. RCS levels associated with Category C EALs are listed in Table C-5.

4. RCS Temperature

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

5. Communications

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

6. Inadvertent Criticality

Inadvertent criticalities pose potential personnel safety hazards as well as being indicative of losses of reactivity control.

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** AC power capability to 480V safeguards buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 480V safeguards bus power

**EAL:**

**CU1.1 Unusual Event**

AC power capability to 480V safeguards buses reduced to a single power source, Table C-1, for ≥ 15 min. (Note 4)

**AND**

**Any** additional single power source failure will result in a complete loss of all 480V safeguards bus power

Note 4: The ED 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.

| Table C-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Safeguard train A, Buses 14 & 18)<br>• EDG 1B (Safeguard train B, Buses 16 & 17) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel, D - Defueled

**Basis:**

<u>Generic</u>

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 480V safeguards bus AC power. This condition could occur due to a loss of off-site power with a concurrent failure of one emergency generator to supply power to its emergency bus. The subsequent loss of this single power source would escalate the event to an Alert in accordance with EAL CA1.1.

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

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., charging pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are specifically included in Table C-1 because service water pump operation is necessary for decay heat removal while in cold conditions.

There are three offsite power sources available to these buses (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and an Unusual Event must be declared.

There are two onsite emergency AC power sources available in the cold modes:

- EDG 1A
- EDG 1B

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to be capable of supplying one or more safety-related buses within

15 minutes, an Unusual Event is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to an Alert under EAL CA1.1.

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. NEI 99-01 CU3

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** Loss of **all** offsite and **all** onsite AC power to 480V safeguards buses for ≥ 15 min.

**EAL:**

**CA1.1 Alert**

Loss of **all** offsite and **all** onsite AC power, Table C-1, to 480V safeguards buses for ≥ 15 min. (Note 4)

Note 4: The ED 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.

| **Table C-1 AC Power Sources** | |
|---|---|
| **Onsite** | • EDG 1A (Safeguard train A, Buses 14 & 18)<br>• EDG 1B (Safeguard train B, Buses 16 & 17) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel, D - Defueled

**Basis:**

Generic

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink.

The event can be classified as an Alert when in cold shutdown, refueling, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by EALs in Category R.

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

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., charging pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are specifically included in Table C-1 because service water pump operation is necessary for decay heat removal while in cold conditions.

There are three offsite power sources available to these buses in the cold modes (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T).
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and Alert must be declared.

There are two onsite emergency AC power sources available in the cold modes:

- EDG 1A
- EDG 1B

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If all sources fail to be capable of supplying all safety-related buses within 15 minutes, an Alert is declared under this EAL.

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. NEI 99-01 CA3

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – Loss of DC Power

**Initiating Condition:** Loss of **required** DC power for ≥ 15 min.

**EAL:**

| | |
|---|---|
| **CU2.1** | **Unusual Event** |
| < 108 VDC on **required** 125 VDC buses for ≥ 15 min. (Note 4) | |

Note 4: The ED 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.

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations.

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

Plant-Specific

The 125 VDC vital system is divided into two independent and isolated channels. Each channel consists of one battery, two battery chargers, one DC bus and one inverter. Each inverter has an associated vital AC distribution panel board. Power to the DC bus, DC unit control panels, and inverters is supplied by the station batteries and/or the battery chargers. Each battery charger is fully rated and can recharge a discharged battery while at the same time supplying the steady state power requirements of the system.

A separate TSC Battery system is designed with an intertie to each of the two main (A and B) distribution panels for use during maintenance and abnormal conditions.

The safety-related station batteries have been sized to carry their expected shutdown loads following a plant trip and loss of offsite power or following a station blackout without battery terminal voltage falling below 108.6 volts for a period of 4 hours (ref. 1).

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

The loss of the TSC Battery does not constitute an entry condition for this EAL.

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS2.1.

**Ginna Basis Reference(s):**

1. UFSAR Section 8.3.2 Direct Current Power Systems
2. O-6.13 Daily Surveillance Log
3. NEI 99-01 CU7

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** RCS leakage

**EAL:**

| | |
|---|---|
| **CU3.1** | **Unusual Event** |
| RCS leakage results in the inability to maintain or restore RCS level within the target band established by procedure for ≥ 15 min. (Note 4) | |

Note 4: The ED 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

**Mode Applicability:**

5 - Cold Shutdown

**Basis:**

Generic

This EAL is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

Prolonged loss of RCS inventory may result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL CA3.1.

Plant-Specific

This EAL is applicable if RCS level cannot be restored and maintained within the prescribed target band specified by procedure.

**Ginna Basis Reference(s):**

2. AP-RCS.1, Reactor Coolant Leak
3. NEI 99-01 CU1

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** RCS Leakage

**EAL:**

**CU3.2 Unusual Event**

Unplanned RCS level drop below **EITHER** of the following for ≥ 15 min. (Note 4):

Reactor Vessel flange (84 in. on loop level indicators) (when the level band is established above the flange)

**OR**

RCS level target band established by procedure (when the level band is established below the flange)

Note 4: The ED 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

**Mode Applicability:**

6 - Refuel

**Basis:**

Generic

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An unplanned event that results in water level decreasing below the Reactor Vessel flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the Reactor Vessel flange), warrants declaration of a UE due to the reduced RCS inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL CA3.1.

This EAL involves a decrease in RCS level below the top of the Reactor Vessel flange that continues for 15 minutes due to an unplanned event. This EAL is not applicable to decreases in flooded reactor cavity level, which is addressed by EAL RU2.1, until such time as the level decreases to the level of the vessel flange.

Plant-Specific

The Reactor Vessel flange level (uncorrected) is at 84 in. (252' 6" ele.) on Loop A & B Level Indicators (LIT-432A and B) (ref. 1, 2).

This EAL involves a lowering in RCS level below the top of the Reactor Vessel flange, or the inability to maintain water level above the intended level when level is being intentionally maintained below the flange, that continues for fifteen minutes due to an unplanned event. This EAL is not applicable to drops in flooded refueling pool level (covered by lowering spent fuel pool water level in EAL RU2.1) until such time as the level lowers to the level of the vessel flange. If level continues to lower and reaches the bottom of the RCS Hot Leg reference level (0 in. indicated), escalation to the Alert level under EAL CA3.1 would be appropriate. If the level lowering is accompanied by RCS heatup, escalation to the Alert level under EAL CA4.1 may also be appropriate.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refuel mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refuel mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. Reactor Vessel water level is normally monitored in Refuel mode using the following instruments (ref. 3,5,6):

- Loop A Level Indicator LIT-432A
- Loop B Level Indicator LIT-432B
- Loop B Sightglass
- Cavity Water Level

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. RF-601 Fuel Handling Accident Instructions
2. O-2.3 Draining the Reactor Coolant System to < 84" but > 64"
3. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
4. NEI 99-01 CU2
5. O-15.1 Administrative Requirement Checklist for Entry to Mode 6 and Refueling Conditions
6. O-6.13 Daily Surveillance Log

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** RCS Leakage

**EAL:**

**CU3.3 Unusual Event**

RCS level **cannot** be monitored with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage

| **Table C-2 RCS Leakage Indications** |
|---|
| • Containment Sump A |
| • Containment Sump B |
| • Auxiliary Building Sump Tank |
| • Reactor Coolant Drain Tank (RCDT) |

**Mode Applicability:**

6 - Refuel

**Basis:**

<u>Generic</u>

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the Reactor Vessel flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the Reactor Vessel flange), warrants declaration of a UE due to the reduced RCS inventory that is available to keep the core covered.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA3.1 or EAL CA4.1.

This EAL addresses conditions in the Refuel mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of RCS level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RCS inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Plant-Specific

In this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by Containment Sumps, Auxiliary Building Sump or RCDT level changes (ref. 1, 2). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

**Ginna Basis Reference(s):**

1. UFSAR 5.1.3.6 Design Criteria
2. UFSAR 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
3. NEI 99-01 CU2

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** Loss of RCS inventory

**EAL:**

**CA3.1 Alert**

Loss of inventory as indicated by RCS water level < 0 in.

**OR**

RCS level **cannot** be monitored for ≥ 15 min. with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage (Note 4)

Note 4: The ED 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.

| Table C-2 RCS Leakage Indications |
|---|
| • Containment Sump A |
| • Containment Sump B |
| • Auxiliary Building Sump Tank |
| • Reactor Coolant Drain Tank (RCDT) |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS level decrease and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert.

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

If RCS level continues to lower then escalation to Site Area Emergency will be via EAL CS3.1, EAL CS3.2 or EAL CS3.3.

<u>Plant-Specific</u>

When RCS water level lowers to 0 in. (uncorrected) on loop level indicators, the bottom of the RCS hot leg level instrument tap is uncovered (ref. 1, 2). This level can be monitored by:

- Loop A Level Indicator LIT-432A
- Loop B Level Indicator LIT-432B
- Loop B Sightglass

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel level lowering and potential core uncovery. The bottom of the hot leg is the level equal to the bottom of the Reactor Vessel loop penetration, not the low point of the loop. This level was chosen because remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The inability to restore and maintain level after reaching this setpoint implies a failure of the RCS barrier.

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

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refuel mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refuel mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed

(Cavity level monitoring with the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted (ref. 7,8).

In the second condition of this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by Containment Sumps, Auxiliary Building Sump or RCDT level changes (ref. 1, 2, 3, 5). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency EAL duration. The interval allows this EAL to be an effective precursor to the Site Area Emergency EAL CS3.1. Therefore this EAL meets the definition for an Alert emergency.

**Ginna Basis Reference(s):**

1. RF-601 Fuel Handling Accident Instructions
2. O-2.3 Draining the Reactor Coolant System to < 84" but > 64"
3. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
4. UFSAR 5.1.3.6 Monitoring Reactor Coolant Leakage
5. UFSAR 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
6. NEI 99-01 CA1
7. O-15.1 Administrative Requirement Checklist for Entry to Mode 6 and Refueling Conditions
8. O-6.13 Daily Surveillance Log

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

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

**EAL:**

**CS3.1 Site Area Emergency**

RCS level **cannot** be monitored with a loss of RCS inventory as indicated by **ANY** of the following for ≥ 30 min. (Note 4):

- Containment radiation R-29 or R-30 > 1.0E+02 R/hr
- Erratic Source Range Nuclear Instrumentation indication
- Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage

Note 4: The ED 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.

| Table C-2 RCS Leakage Indications |
|---|
| • Containment Sump A |
| • Containment Sump B |
| • Auxiliary Building Sump Tank |
| • Reactor Coolant Drain Tank (RCDT) |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

<u>Generic</u>

Under the conditions specified by this EAL, continued decrease in RCS level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RCS. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, RG1.1, RG1.2 or RG1.3.

The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

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 site specific monitor indication and possible alarm.

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

Plant-Specific

In Refuel or Cold Shutdown mode, normal RCS level indication may be unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

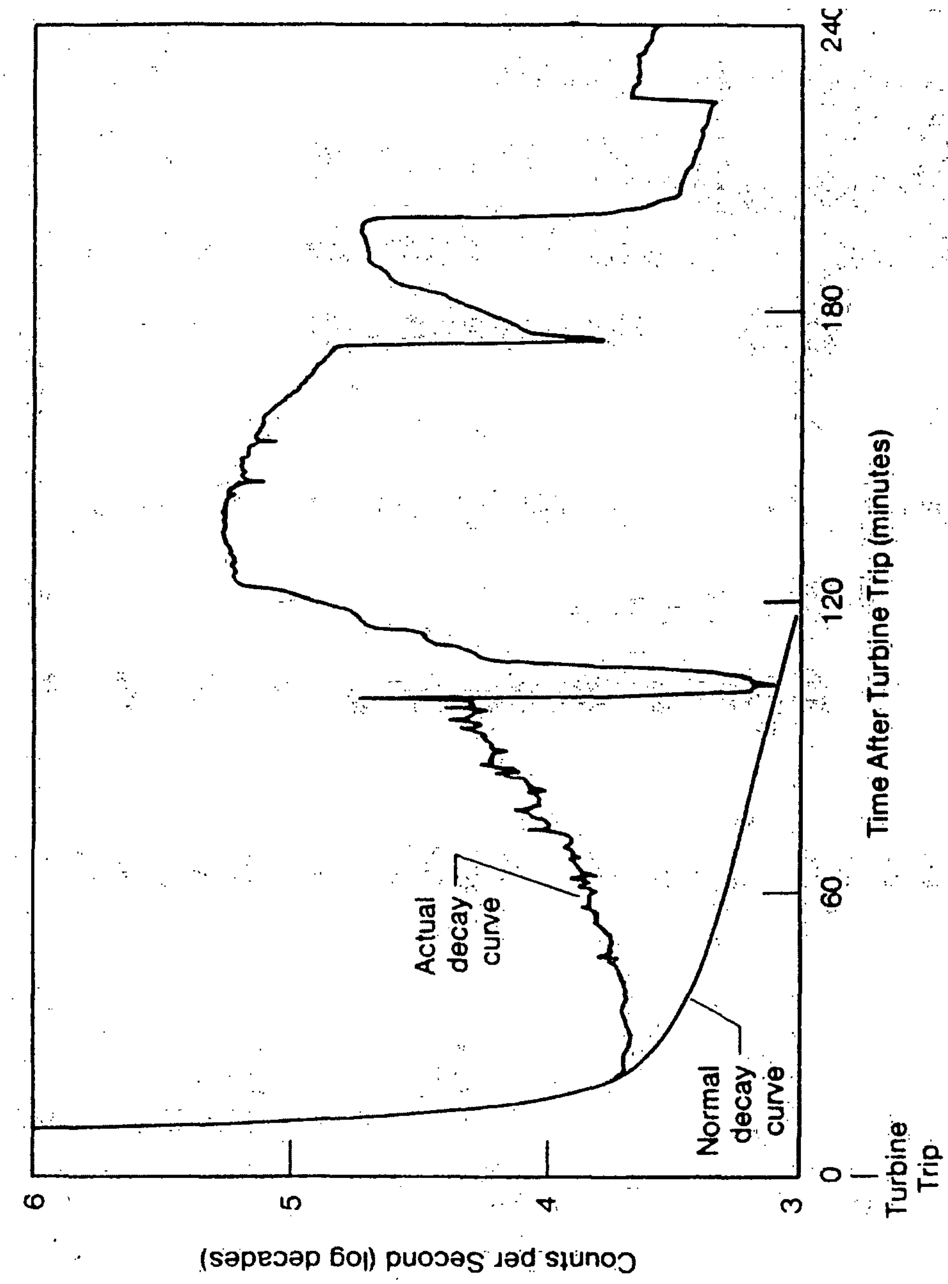
- As water level in the Reactor Vessel lowers, the dose rate above the core will rise. Containment radiation is indicated on R-29 and R-30. The dose rate due to this core shine should result in on-scale Containment radiation monitor indication and possible alarm. Assuming total draindown of the upper cavity, line-of-sight dose rates from a fully exposed upper internal package would be approximately 300 R/hr. The containment radiation monitors high alarm is set at 1.0E+02 R/hr (ref 1). The 1.0E+02 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions while in the Refuel mode.
- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors such as Source Range Nuclear Instrumentation N-31 and N-32 can be used as a tool for making such determinations (ref 2). Figure C-1 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor.
- If water level indication is unavailable, Reactor Vessel inventory loss must be detected by Containment Sumps, Auxiliary Building Sump or RCDT level changes

(ref. 3, 4, 5, 6, 7). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. P-6 Precautions, Limitations and Setpoints Nuclear Instrumentation System
3. RF-601 Fuel Handling Accident Instructions
4. O-2.3 Draining the Reactor Coolant System to < 84" but > 64"
5. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
6. UFSAR 5.1.3.6 Monitoring Reactor Coolant Leakage
7. UFSAR 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
8. NEI 99-01 CS1

**Figure C-1: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident**



**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Level

**Initiating Condition:** Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged

**EAL:**

**CG3.1 General Emergency**

RCS level **cannot** be monitored with core uncovery indicated by **ANY** of the following for ≥ 30 min. (Note 4):

- Containment radiation R-29 or R-30 > 1.0E+02 R/hr
- Erratic Source Range Nuclear Instrumentation indication
- Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage

**AND**

**Any** Containment Challenge Indication, Table C-3

Note 4: The ED 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

| **Table C-2 RCS Leakage Indications** |
|---|
| • Containment Sump A<br>• Containment Sump B<br>• Auxiliary Building Sump Tank<br>• Reactor Coolant Drain Tank (RCDT) |

| **Table C-3 Containment Challenge Indications** |
|---|
| • Containment closure **not** established<br>• Hydrogen concentration in Containment ≥ 4%<br>• Unplanned rise in Containment pressure |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This EAL represents the inability to restore and maintain RCS level to above the top of active fuel with containment challenged. Fuel damage is probable if RCS level cannot be restored, as available decay heat will cause boiling, further reducing the RCS level. With the Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers.

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

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

If Containment Closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to General Emergency would not occur.

Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

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

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

Plant-Specific

In Refuel or Cold Shutdown mode, normal RCS level indication may be unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- As water level in the Reactor Vessel lowers, the dose rate above the core will rise. Containment radiation is indicated on R-29 and R-30. The dose rate due to this core shine should result in on-scale Containment radiation monitor indication and possible alarm. Assuming total draindown of the upper cavity, line-of-sight dose rates from a fully exposed upper internal package would be approximately 300 R/hr. The containment radiation monitors high alarm is set at 1.0E+02 R/hr (ref 1). The 1.0E+02 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions while in the Refuel mode.

- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors such as Source Range Nuclear Instrumentation N-31 and N-32 can be used as a tool for making such determinations (ref 2). Figure C-1 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor.
- If water level indication is unavailable, Reactor Vessel inventory loss must be detected by Containment Sumps, Auxiliary Building Sump or RCDT level changes (ref. 3, 4, 5, 6, 7). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Three indications are associated with Containment challenges:

- Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in O-2.3.1A "Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation", provide a functional barrier to fission product release (ref 8). Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires that, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS bulk boiling.

- 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 Containment. However, Containment monitoring (CH-EPIP-CVH2) and/or sampling should be performed to verify this assumption. A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4.1% (rounded to 4%) by volume (ref. 9).
- Unplanned Containment pressure increases are not expected during Cold Shutdown or Refuel mode. The threshold is indicative of conditions challenging containment closure.

Definitions:

**Containment Closure**

The site-specific procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to Ginna, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in O-2.3.1A, provide a functional barrier to fission product release.

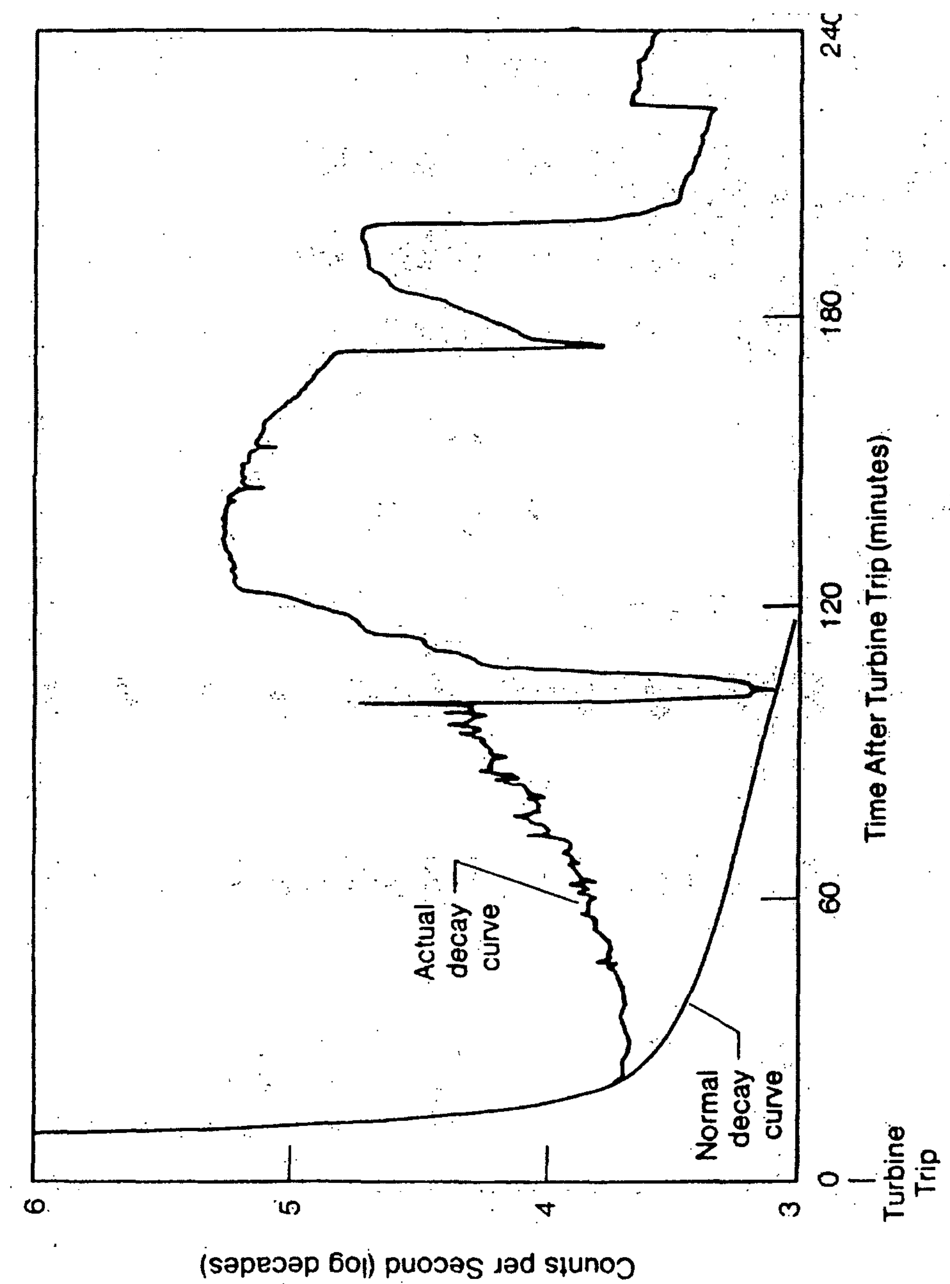
**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. P-6 Precautions, Limitations and Setpoints Nuclear Instrumentation System
3. RF-601 Fuel Handling Accident Instructions
4. O-2.3 Draining the Reactor Coolant System to < 84" but > 64"
5. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
6. UFSAR 5.1.3.6 Monitoring Reactor Coolant Leakage
7. UFSAR 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
8. O-2.3.1A Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation
9. SACRG-1 Severe Accident Control Room Guideline Initial Response
10. NEI 99-01 CG1

**Figure C-1: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident**



**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 4 – RCS Temperature

**Initiating Condition:** Unplanned loss of decay heat removal capability

**EAL:**

| | |
|---|---|
| **CU4.1** | **Unusual Event** |
| Unplanned event results in RCS temperature > 200°F | |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This EAL is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered.

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

Normal means of core temperature indication and RCS level indication may not be available in the Refuel mode. Redundant means of RCS level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1 based on exceeding its temperature duration or pressure criteria.

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F) (ref. 1). These include (ref. 2):

- The average of T0409A ($T_{HOT}$) and T0410B ($T_{COLD}$) for forced circulation with A RCP pump running

- The average T0410B ($T_{COLD}$) **AND** either T0410A **OR** incore thermocouples for $T_{HOT}$ for forced circulation with B RCP pump running
- T0630 Residual Heat Removal Pump Discharge Header
- Incore Temperatures

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. Technical Specifications Table 1.1-1
2. O-2.2 Plant Shutdown from Hot Shutdown to Cold Conditions
3. NEI 99-01 CU4

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 4 – RCS Temperature

**Initiating Condition:** Unplanned loss of decay heat removal capability

**EAL:**

| | |
|---|---|
| **CU4.2** | **Unusual Event** |
| Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 4) | |

Note 4: The ED 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

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This EAL is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered.

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

Normal means of core temperature indication and RCS level indication may not be available in the Refuel mode. Redundant means of RCS level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown or Refuel modes, this EAL would result in declaration of a UE if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1 based on exceeding its temperature criteria.

Plant-Specific

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refuel mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refuel mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. Reactor Vessel water level is normally monitored in Refuel mode using the following instruments (ref. 1,5,6):

- Loop A Level Indicator LIT-432A
- Loop B Level Indicator LIT-432B
- Loop B Sightglass
- Cavity Water Level

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F) (ref. 2). These include (ref. 3):

- The average of T0409A ($T_{HOT}$) and T0410B ($T_{COLD}$) for forced circulation with A RCP pump running
- The average of T0410B ($T_{COLD}$) **AND** either T0410A **OR** incore thermocouples for $T_{HOT}$ for forced circulation with B RCP pump running
- T0630 Residual Heat Removal Pump Discharge Header
- Incore Temperatures

**Ginna Basis Reference(s):**

1. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
2. Technical Specifications Table 1.1-1
3. O-2.2 Plant Shutdown from Hot Shutdown to Cold Conditions
4. NEI 99-01 CU4
5. O-15.1 Administrative Requirement Checklist for Entry to Mode 6 and Refueling Conditions
6. O-6.13 Daily Surveillance Log

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 4 – RCS Temperature

**Initiating Condition:** Inability to maintain plant in cold shutdown

**EAL:**

**CA4.1 Alert**

An unplanned event results in **EITHER**:

RCS temperature > 200°F for > Table C-4 duration

**OR**

RCS pressure increase > 10 psi due to an unplanned loss of decay heat removal capability (this condition is **not** applicable in solid plant conditions)

| Table C-4 RCS Reheat Duration Thresholds | | |
|---|---|---|
| RCS Status | Containment Closure Status | Duration |
| Intact **AND not** reduced inventory | N/A | 60 min.* |
| **Not** intact **OR** reduced inventory | Established | 20 min.* |
| | **Not** established | 0 min. |

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

**Mode Applicability:**

5 - Cold Shutdown, 6 – Refuel

**Basis:**

<u>Generic</u>

The RCS Reheat Duration Thresholds table addresses complete loss of functions required for core cooling for greater than 60 minutes during refuel and cold shutdown modes when RCS integrity is established. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

The RCS Reheat Duration Thresholds table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during refuel and cold shutdown modes when Containment Closure is established but RCS integrity is not established or RCS inventory is reduced. The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible.

Finally, complete loss of functions required for core cooling during refuel and cold shutdown modes when neither Containment Closure nor RCS integrity are established is addressed. No delay time is allowed because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

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

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

Escalation to Site Area Emergency would be via EAL CS3.1 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

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

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F) (ref. 1). These include (ref. 2):

- The average of T0409A ($T_{HOT}$) and T0410B ($T_{COLD}$) for forced circulation with A RCP pump running
- The average of T0410B ($T_{COLD}$) **AND** either T0410A **OR** incore thermocouples for $T_{HOT}$ for forced circulation with B RCP pump running
- T0630 Residual Heat Removal Pump Discharge Header

Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in O-2.3.1A "Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation" (ref. 3), provide a functional barrier to fission product release. Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS boiling.

Reduced Inventory (administrative) is defined as RCS level less than 64 in. on the RCS Loop indicators (ref. 4).

The pressure rise of greater than 10 psig implies an RCS temperature in excess of the Technical Specification cold shutdown limit (200°F) for which this EAL would otherwise permit up to sixty minutes to restore RCS cooling before declaration of an Alert (RCS intact). This EAL therefore covers situations in which it is determined that, due to high decay heat loads, the time provided to reestablish temperature control should be less than sixty minutes (as indicated by significant RCS re-pressurization).

Pressure indicator PI-420 Rx Clnt Loop Lo Rng Press is capable of measuring pressure changes of 10 psig. This represents the visual resolution of the device, with the smallest scale increment of 10 psig (Basis: Walkdown). Escalation to a Site Area Emergency would be under EAL CS2.1 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

Definitions:

**Containment Closure**

The site-specific procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to Ginna, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in O-2.3.1A, provide a functional barrier to fission product release.

**RCS Intact**

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

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. Technical Specifications Table 1.1-1
2. O-2.2 Plant Shutdown from Hot Shutdown to Cold Conditions
3. O-2.3.1A Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation
4. O-2.3.1 Draining and Operation at Reduced Inventory of the Reactor Coolant System
5. NEI 99-01 CA4

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 5 – Communications

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

**EAL:**

**CU5.1 Unusual Event**

Loss of **all** Table C-5 onsite (internal) communication methods affecting the ability to perform routine operations

**OR**

Loss of **all** Table C-5 offsite (external) communication methods affecting the ability to perform offsite notifications

| **Table C-5 Communications Systems** | | |
|---|---|---|
| **System** | **Onsite** (internal) | **Offsite** (external) |
| Commercial phone system | X | X |
| Direct Dial POTS Lines (Blue Phones) | X | X |
| Plant Page Party system | X | |
| Radios/Walkie Talkies | X | |
| FTS 2001 telephone system (ENS, HPN) | | X |
| Control Room Hard Wired Satellite Phone | | X |
| Control Room Emergency Cell Phone | | X |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel, D - Defueled

**Basis:**

<u>Generic</u>

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant issues. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to off-site locations, etc.) are being utilized to make communications possible.

Plant-Specific

Onsite/offsite communications systems are listed in Table C-2 (ref. 1, 2).

This EAL is the cold condition equivalent of the hot condition EAL SU6.1.

**Ginna Basis Reference(s):**

1. A-56 Communication Systems at Ginna Station
2. ER-COMM.1 Loss of Communications
3. NEI 99-01 CU6

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 6 – Inadvertent Criticality

**Initiating Condition:** Inadvertent criticality

**EAL:**

| | |
|---|---|
| **CU6.1** | **Unusual Event** |
| An unplanned sustained positive startup rate observed on nuclear instrumentation | |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refuel

**Basis:**

Generic

This EAL addresses criticality events that occur in Cold Shutdown or Refuel modes such as fuel mis-loading events and inadvertent dilution events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification.

Escalation would be by Emergency Director judgment.

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the rise in neutron population due to subcritical multiplication. Short-term positive startup rates can also be due to welding activities.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. NEI 99-01 CU8

**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.

The events of this category pertain to the following subcategories:

1. Natural or Destructive Phenomena

Natural events include hurricanes, earthquakes or tornadoes that have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety. Non-naturally occurring events that can cause damage to plant facilities include aircraft crashes, missile impacts, etc.

2. Fire or Explosion

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

3. Hazardous Gas

Non-naturally occurring events that can cause damage to plant facilities and include toxic, asphyxiant, corrosive or flammable gas leaks.

4. Security

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

5. 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.

6. 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 Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

**HU1.1 Unusual Event**

Seismic event identified by **ANY two** of the following:

- Red LED event indicator on Kinemetrics ETNA Digital Recorder indicates seismic event detected
- Earthquake felt onsite
- National Earthquake Information Center (Note 6)

Note 6: 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 Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude**.

**Mode Applicability:**

All

**Basis:**

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.

As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.

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

Plant-Specific

A strong motion accelerograph is installed in the subbasement of the intermediate building at elevation 237 ft (ref. 1).

Ginna seismic instrumentation actuates upon sensing any ground motion greater than 0.01g. Registration of a tremor > 0.01g is indicated by a red light on the event indicator at the bottom of the accelerograph case (ref. 2, 3, 4).

The National Earthquake Information Center (NEIC) can confirm seismic activity in the vicinity of Ginna. 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 R. E. Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude** (ref. 5).

**Ginna Basis Reference(s):**

1. UFSAR Section 3.7.4 Seismic Instrumentation
2. ER-SC.4 Earthquake Emergency Plan
3. CPI-ACCELEROGRAPH-51 Functional Check of Kinemetrics Strong Motion Accelerograph
4. VTD-K3356-4104 Kinematics, ETNA Strong Motion Accelerograph Schematics
5. USAR Section 2.1.1 Site Location and Description
6. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

**HU1.2 Unusual Event**

Tornado striking within Protected Area boundary

**OR**

Sustained high winds > 75 mph

**Mode Applicability:**

All

**Basis:**

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

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

Escalation of this emergency classification level, if appropriate, would be based on visible damage, or by other in plant conditions, via EAL HA1.2.

Plant-Specific

All Class 1 structures are designed for a wind velocity of 75 mph assuming FSAR "severe environmental loading" conditions (ref. 1).

Wind speed can be measured up to 100 mph on the 250' and 150' wind speed recorder 'A' (ref. 2). Sustained winds are the five-minute average wind speed.

The Protected Area Boundary is depicted in Drawing 33013-2722 (ref. 3)

Definitions:

**Protected Area**

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

**Ginna Basis Reference(s):**

1. UFSAR Section 3.3.2.1.4 Wind and Tornado Loadings - Input Load Criteria
2. CPI-MET-250 Calibration of Ginna Station Meteorological Wind Speed and Wind Direction Translator Cards
3. Drawing 33013-2722 Residential AC Power Distribution Circuit - Site Layout
4. ER-SC.1 Adverse Weather Plan
5. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

**HU1.3 Unusual Event**

Internal flooding that has the potential to affect **ANY** safety-related structure, system, or component required by Technical Specifications for the current operating mode in **ANY** Table H-1 area

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

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

Escalation of this emergency classification level, if appropriate, would be based visible damage via EAL HA1.3, or by other plant conditions.

Plant-Specific

This threshold addresses the effect of flooding caused by internal events such as component failures, Circulating, Component Cooling or Service Water line ruptures, equipment misalignment, fire suppression system actuation, and outage activity mishaps.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of its removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

This threshold addresses events that may have resulted in a Safe Shutdown Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Safe Shutdown Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern (ref. 1, 2).

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

| | |
|---|---|
| **HU1.4** | **Unusual Event** |
| Turbine failure resulting in casing penetration or damage to turbine or generator seals | |

**Mode Applicability:**

All

**Basis:**

Generic

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

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.

Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via EAL HU2.1 and EAL HU3.1.

This EAL is consistent with the definition of a UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of this emergency classification level, if appropriate, would be to EAL HA1.4 based on damage done by PROJECTILES generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the Category R EALs or Category F EALs.

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external projectiles will be released. These ejected projectiles may impact various plant structures, including those housing safety related equipment.

Failure of turbine or generator seals may be indicated by a loss of seal oil pressure or loss of condenser vacuum (ref. 1).

**Ginna Basis Reference(s):**

1. ER-SC.8 Turbine Blade Failure and Missile Emergency Plan
2. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

**HU1.5 Unusual Event**

Deer Creek flooding over entrance road bridge hand rail

**OR**

Lake level > 252 ft

**OR**

Screen House Suction Bay water level < 17 ft or < 15.5 ft by manual level measurement

**Mode Applicability:**

All

**Basis:**

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses other site specific phenomena that can also be precursors of more serious events.

Plant-Specific

This threshold addresses high and low lake water level conditions that could be a precursor of more serious events.

Ginna plant grade is generally at 270 ft mean sea level (msl) except the area between the lake and Turbine Building which is at grade 253 ft msl. Lake water level > 253.28 ft msl corresponds to plant design levels (ref. 1). A lake level of 252 ft has been selected for this threshold to be anticipatory of exceeding design flood levels and is the level at which flood control actions are procedurally taken (ref. 2).

Flooding in Deer Creek above the plant entrance handrails will ultimately result in water accumulation in the Turbine Building and Screenhouse (ref. 2). This may preclude emergency response personnel access and egress.

High lake level may be determined using markers attached to a metal pole mounted on the discharge canal bridge upstream of the submarine net. The high level markers are at lake levels of 252 ft and 253 ft (ref. 2).

The Screenhouse Lo-Lo level alarm actuates at 19' indicated (ref. 3). When Screenhouse Suction Bay water level drops to 17.0 ft indicated (this corresponds to a level of 15.5' measured manually) increased Control Room monitoring is initiated. This level has been selected for this threshold to be anticipatory of a potential loss of service water system pump suction at 16.0 ft (ref. 4).

**Ginna Basis Reference(s):**

1. UFSAR Section 3.4.1 Flood Protection
2. ER-SC.2 High Water (Flood) Plan
3. AR-I-9 Screen House Lo-Lo Level 19'
4. ER-SC.3 Low Screenhouse Water Level
5. NEI 99-01 HU1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.1 Alert**

**EITHER**:

Confirmation of an earthquake of an intensity > 0.08 g per ER-SC.4 Earthquake Emergency Plan

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component

**AND**

Earthquake confirmed by **EITHER**:

Earthquake felt in onsite

**OR**

National Earthquake Information Center (Note 6)

Note 6: 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 Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude**.

**Mode Applicability:**

All

**Basis:**

Generic

These EALs escalate from HU1.1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

Seismic events of this magnitude can result in a vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

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

Plant-Specific

This EAL is based on the UFSAR design basis operating earthquake of 0.08 g acceleration (ref. 1). Seismic events of this magnitude can cause damage to plant safety functions.

Ginna seismic instrumentation actuates upon sensing any seismic activity (ref. 2, 3, 4).

The method of detection of an earthquake greater than OBE intensity relies on either:

- analysis of the Ginna strong motion accelerograph located in the Intermediate Building by plant I&C and plant Engineering (ref. 3)

  **OR**

- by actual indications of degraded safe shutdown system performance

confirmed by either shift operators on duty in the Control Room determining that ground motion was felt, or corroborated by the NEIC.

The National Earthquake Information Center (NEIC) can confirm seismic activity in the vicinity of Ginna. 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 R. E. Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude** (ref. 5).

Definitions:

**Safety-Related Structures, Systems and Components** (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.

**Ginna Basis Reference(s):**

1. UFSAR Section 3.7.1.3 Design Response Spectra
2. ER-SC.4 Earthquake Emergency Plan
3. CPI-ACCELEROGRAPH-51 Functional Check of Kinematics Strong Motion Accelerograph
4. VTD-K3356-4104 ETNA Strong Motion Accelerograph Schematics
5. USAR Section 2.1.1 Site Location and Description
6. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.2 Alert**

Tornado striking or sustained high winds > 75 mph resulting in **EITHER**:

Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL escalates from HU1.2 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL is based on a tornado striking (touching down) or high winds that have caused visible damage to structures containing functions or systems required for safe shutdown of the plant.

Plant-Specific

All Class 1 structures are designed for a wind velocity of 75 mph assuming FSAR "severe environmental loading" conditions (ref. 1). This EAL is based on the structural design basis of 75 mph or impact by tornado. Wind loads of this magnitude can cause damage to safety functions.

Wind speed can be measured up to 100 mph on the 250' and 150' wind speed recorder 'A' (ref. 2). Sustained winds are the five-minute average wind speed.

The Protected Area Boundary is depicted in Drawing 33013-2722 (ref. 3).

This threshold addresses events that may have resulted in a Safe Shutdown Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Safe Shutdown Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern. The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Safe Shutdown Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold.

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 4, 5).

Definitions:

**Visible Damage**

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

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. UFSAR Section 3.3.2.1.4 Wind and Tornado Loadings - Input Load Criteria
2. CPI-MET-250 Calibration of Ginna Station Meteorological Wind Speed and Wind Direction Translator Cards
3. Drawing 33013-2722 Residential AC Power Distribution Circuit - Site Layout
4. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
5. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
6. ER-SC.1 Adverse Weather Plan
7. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.3** **Alert**

Internal flooding in **ANY** Table H-1 area resulting in **EITHER**:

An electrical shock hazard that precludes access to operate or monitor **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| **Table H-1 Safe Shutdown Areas** |
| --- |
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

Plant-Specific

This threshold addresses the effect of flooding caused by internal events such as component failures such as Circulating, Component Cooling or Service Water line ruptures, equipment misalignment, fire suppression system actuation, steam leaks or outage activity mishaps.

Safe Shutdown Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern (ref. 1, 2).

Uncontrolled internal flooding that has degraded safety shutdown equipment or created a safety hazard precluding access necessary for the safe operation or monitoring of safety equipment warrants declaration of an Alert.

Definitions:

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.4 Alert**

Turbine failure-generated projectiles resulting in **EITHER**:

Visible damage to or penetration of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL escalates from HU1.4 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

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

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external projectiles will be released. These ejected projectiles may impact various plant structures, including those housing safety related equipment.

Failure of turbine or generator seals may be indicated by a loss of seal oil pressure or loss of condenser vacuum (ref. 1).

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 2, 3).

Definitions:

**Projectile**

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

**Visible Damage**

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

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. ER.SC-8 Turbine Blade Failure and Missile Emergency Plan
2. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
3. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
4. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.5 Alert**

Lake level > 253 ft

**OR**

Screen House Suction Bay water level < 16 ft or < 14.5 ft by manual level measurement

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses other site specific phenomena that result in visible damage to vital areas or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant that can also be precursors of more serious events.

Plant-Specific

This threshold covers high and low water level conditions that may have resulted in a plant safe shutdown area being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

Ginna plant grade is generally at 270 ft mean sea level (msl) except the area between the lake and Turbine Building which is at grade 253 ft msl. Lake water level > 253.28 ft msl corresponds to plant design levels (ref. 1). A lake level of 253 ft has been selected for this threshold to be indicative of exceeding design flood levels (ref. 2).

High lake level may be determined using markers attached to a metal pole mounted on the discharge canal bridge upstream of the submarine net. The high level markers are at lake levels of 252 ft and 253 ft (ref. 2).

The Screenhouse Lo-Lo level alarm actuates at 19' indicated (ref. 3). If indicated service water pump bay level drops below 16 ft (this corresponds to a lake level of 14.5' measured manually) the service water pumps are declared inoperable. This level has been selected for this threshold to be indicative of a loss of service water system pump suction (ref. 4).

**Ginna Basis Reference(s):**

1. UFSAR Section 3.4.1 Flood Protection
2. ER-SC.2 High Water (Flood) Plan
3. AR-I-9 Screen House Lo-Lo Level 19'
4. ER-SC.3 Low Screenhouse Water Level
5. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.6 Alert**

Vehicle crash resulting in **EITHER**:

Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses vehicle crashes within the Protected Area that results in visible damage to vital areas or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.

Plant-Specific

This EAL is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. Vehicle types include automobiles, aircraft, trucks, cranes, forklifts, waterborne craft, etc.

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 1, 2).

Definitions:

**Visible Damage**

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

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HA1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire within the Protected Area **not** extinguished within 15 min. of detection or explosion within the Protected Area

**EAL:**

**HU2.1 Unusual Event**

Fire **not** extinguished within 15 min. of Control Room notification or verification of a Control Room fire alarm in **ANY** Table H-1 area or Turbine Building (Note 4)

Note 4: The ED 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.

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses the magnitude and extent of fires that may be potentially significant precursors of damage to safety systems. It addresses the FIRE, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and report by plant personnel or sensor alarm indication.

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems.

As used here, notification is visual observation and report by plant personnel or sensor alarm indication.

The 15-minute period to extinguish the fire begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm.

Determination of a valid fire detection system alarm includes actions that can be taken within the Control Room or at nearby Fire Panels to determine that the alarm is not spurious. These actions include the use of direct or indirect indications such as redundant alarms or instrumentation readings associated with the area to ensure the alarm is not spurious and is an indication of a fire. An alarm verified in this manner is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm. If the alarm cannot be verified by redundant Control Room or nearby Fire Panel indications, notification from the field that a fire exists would be required to start the 15-minute classification and fire extinguishment clocks.

The intent of this 15 minute duration is to size the fire and to discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket).

Plant-Specific

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 1, 2). The Turbine Building is included because it is immediately adjacent to one or more Table H-1 areas and a fire within the Turbine Building may potentially impact safe shutdown equipment should the fire not be controlled.

Definitions:

**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.

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HU2

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire within the Protected Area **not** extinguished within 15 min. of detection or explosion within the Protected Area

**EAL:**

| **HU2.2** | **Unusual Event** |
|---|---|
| Explosion within the Protected Area | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses the magnitude and extent of explosions that may be potentially significant precursors of damage to safety systems. It addresses the explosion, and not the degradation in performance of affected systems that may result.

This EAL addresses only those explosions of sufficient force to damage permanent structures or equipment within the Protected Area.

No attempt is made to assess the actual magnitude of the damage. The occurrence of the explosion is sufficient for declaration.

The Emergency Director also needs to consider any security aspects of the explosion, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA2.1.

Plant-Specific

While some explosions may also result in fires that exceed EAL HU2.1, no fire is necessary to declare an emergency in the event of an explosion. If a fire also occurs as a result or with an explosion, declare the Unusual Event based on the explosion and monitor the progress of the fire for potential escalation due to fire damage.

Definitions:

**Explosion**

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

**Protected Area**

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

**Ginna Basis Reference(s):**

1. Drawing 33013-2722 Residential AC Power Distribution Circuit - Site Layout
2. NEI 99-01 HU2

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown

**EAL:**

**HA2.1 Alert**

Fire or explosion resulting in **EITHER**:

Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

**OR**

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Containment Building |
| • Auxiliary Building |
| • Control Building |
| • Intermediate Building |
| • Emergency Diesel Building(s) |
| • SAFW Building |
| • Screenhouse |
| • Cable Tunnel |
| • Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

Visible damage is used to identify the magnitude of the fire or explosion and to discriminate against minor fires and explosions.

The reference to structures containing safety systems or components is included to discriminate against fires or explosions in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the fire or explosion was large enough to cause damage to these systems.

The use of visible damage should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency with the resources needed to perform detailed damage assessments.

The Emergency Director also needs to consider any security aspects of the explosion.

Escalation of this emergency classification level, if appropriate, will be based on EALs in Category S, Category F or Category R.

Plant-Specific

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 1, 2).

Definitions:

**Explosion**

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

**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.

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HA2

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 3 – Hazardous Gas

**Initiating Condition:** Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

**EAL:**

| | |
|---|---|
| **HU3.1** | **Unusual Event** |
| Release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This EAL is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

Plant-Specific

Normal plant operations is defined to mean activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

Definitions:

**Normal Plant Operations**

Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

**Ginna Basis Reference(s):**

1. NEI 99-01 HU3

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 3 – Hazardous Gas

**Initiating Condition:** Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

**EAL:**

| **HU3.2** | **Unusual Event** |
|---|---|
| Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event | |

**Mode Applicability:**

All

**Basis:**

Generic

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

Plant-Specific

None

**Ginna Basis Reference(s):**

1. NEI 99-01 HU3

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 3 – Hazardous Gas

**Initiating Condition:** Access to a Vital Area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor

**EAL:**

**HA3.1 Alert**

Access to **ANY** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of **ANY** safety-related structure, system, or component (Note 5)

Note 5: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should **not** be declared as it will have **no** adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

**Mode Applicability:**

All

**Basis:**

Generic

Gases in a Vital Area can affect the ability to safely operate or safely shutdown the reactor.

The fact that SCBA may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases.

This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If access is not required at the time the unsafe concentrations exist in the affected area or if the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gasses which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on EALs in Category S, Category F or Category R.

Plant-Specific

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 1, 2).

If hazardous gas concentration in a Table H-1 area restricts access but the equipment is not required to be operable or will not be required to operate before access can be reestablished (e.g., fans are ventilating the area), this EAL should not be declared.

Definitions:

**Safety-Related Structures, Systems and Components** (as defined in 10 CFR 50.2)

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.

**Ginna Basis Reference(s):**

1. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
2. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
3. NEI 99-01 HA3

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

**Initiating Condition:** Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant

**EAL:**

| | |
|---|---|
| **HU4.1** | **Unusual Event** |

A security condition that does **not** involve a hostile action as reported by Security Shift Supervision

**OR**

A credible site-specific security threat notification

**OR**

A validated notification from NRC providing information of an aircraft threat

**Mode Applicability:**

All

**Basis:**

Generic

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as hostile actions are classifiable under EAL HA4.1, EAL HS4.1 and EAL HG4.1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the Ginna Safeguards Contingency Plan.

First Condition

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

This threshold is based on the Ginna Safeguards Contingency Plan. The Ginna Safeguards Contingency Plan is based on guidance provided by NEI 03-12.

Second Condition

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the Ginna Safeguards Contingency Plan .

Third Condition

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

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level via EAL HA4.1 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

Plant-Specific

If the Security Shift Supervisor determines that a threat notification is credible, the Security Shift Supervisor will notify the Operations Shift Manager that a "Credible Threat" condition exists for Ginna. Generally, Ginna Security procedures address standard practices for determining credibility. The three main criteria for determining credibility are: technical feasibility, operational feasibility, and resolve. For Ginna, a validated notification delivered by the FBI, NRC or similar agency is treated as credible.

Definitions:

**Airliner/Large Aircraft**

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

**Hostile Action**

An act toward Ginna 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 Ginna. 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).

**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.

**Ginna Basis Reference(s):**

1. Ginna Safeguards Contingency Plan
2. ER-SEC.1 Response to Change in Security Threat Level
3. ER-SEC.2 Response to Intrusion by Adversary
4. ER-SEC.3 Response to Airborne Threat
5. NEI 99-01 HU4

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

**Initiating Condition:** Hostile action within the Owner Controlled Area or airborne attack threat

**EAL:**

| **HA4.1** **Alert** |
|---|
| A hostile action is occurring or has occurred within the Owner Controlled Area as reported by Security Shift Supervision |
| **OR** |
| A validated notification from NRC of an airliner attack threat within 30 min. of the site |

**Mode Applicability:**

All

**Basis:**

Generic

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

First Condition

This condition addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Owner Controlled Area. Those events are adequately addressed by other EALs.

Note that this condition is applicable for any hostile action occurring, or that has occurred, in the Owner Controlled Area.

Second Condition

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

The intent of this condition is to ensure that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organizations (OROs) and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This condition is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Plant-Specific

Definitions:

**Airliner/Large Aircraft**

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

**Hostile Action**

An act toward Ginna 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 Ginna. 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).

**Ginna Basis Reference(s):**

1. Ginna Safeguards Contingency Plan
2. NEI 99-01 HA4

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

**Initiating Condition:** Hostile action within the Protected Area

**EAL:**

| | |
|---|---|
| **HS4.1** | **Site Area Emergency** |
| A hostile action is occurring or has occurred within the Protected Area as reported by Security Shift Supervision | |

**Mode Applicability:**

All

**Basis:**

Generic

This condition represents an escalated threat to plant safety above that contained in the Alert in that a hostile force has progressed from the Owner Controlled Area to the Protected Area.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires Offsite Response Organization (ORO) readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Protected Area. Those events are adequately addressed by other EALs.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

Plant-Specific

None

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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**

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

**Ginna Basis Reference(s):**

1. Ginna Safeguards Contingency Plan
2. NEI 99-01 HS4

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 4 – Security

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

**EAL:**

| | |
|---|---|
| **HG4.1** | **General Emergency** |
| A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL encompasses conditions under which a hostile action has resulted in a loss of physical control of Vital Areas (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

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

Plant-Specific

Safety functions include:

- Reactivity control
- RCS Inventory
- Secondary Heat Removal

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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).

**Ginna Basis Reference(s):**

1. NEI 99-01 HG1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety
**Subcategory:** 4 – Security
**Initiating Condition:** Hostile action resulting in loss of physical control of the facility
**EAL:**

| | |
|---|---|
| **HG4.2** | **General Emergency** |
| A hostile action has caused failure of Spent Fuel Cooling systems | |
| **AND** | |
| Imminent fuel damage is likely | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses failure of spent fuel cooling systems as a result of hostile action if imminent fuel damage is likely.

Plant-Specific

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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**

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

**Ginna Basis Reference(s):**

1. NEI 99-01 HG1

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 5 – Control Room Evacuation

**Initiating Condition:** Control Room evacuation has been initiated

**EAL:**

| **HA5.1** | **Alert** |
|---|---|
| Control Room evacuation has been initiated | |

**Mode Applicability:**

All

**Basis:**

Generic

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

Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

Plant-Specific

AP-CR.1 Control Room Inaccessibility provides specific instructions for evacuating the Control Room and establishing plant control in alternate locations (ref. 1).

**Ginna Basis Reference(s):**

1. AP-CR.1 Control Room Inaccessibility
2. NEI 99-01 HA5

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 5 – Control Room Evacuation

**Initiating Condition:** Control Room evacuation has been initiated and plant control **cannot** be established

**EAL:**

| | |
|---|---|
| **HS5.1** | **Site Area Emergency** |
| Control Room evacuation has been initiated | |
| **AND** | |
| Control of the plant **cannot** be established within 30 min. | |

**Mode Applicability:**

All

**Basis:**

Generic

The intent of this EAL is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated).

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink).

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

Escalation of this emergency classification level, if appropriate, would be by EALs in Category F or Category R.

Plant-Specific

AP-CR.1 Control Room Inaccessibility provides specific instructions for evacuating the Control Room and establishing plant control in alternate locations (ref. 1).

An analysis was performed as part of the Fire Protection Program (ref. 2) to determine how quickly control must be re-established at Ginna without core uncovery or damage. There are 5 time-critical actions which must be accomplished to enable established performance goals to be met. In evaluating a reasonable timeline for completion of tasks required in the ER-FIRE procedures to restore charging, it was estimated that restoration should be completed in less than 30 minutes. This is consistent with information obtained during operator walk-throughs of the ER-FIRE procedures which consistently indicated restoration in 17 to 24 minutes.

**Ginna Basis Reference(s):**

1. AP-CR.1 Control Room Inaccessibility
2. Fire Protection Program, Section 3.2.2.12 Time Criteria for Achieving Hot Shutdown
3. NEI 99-01 HS2

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions existing that in the judgment of the Emergency Director warrant declaration of a UE

**EAL:**

| | |
|---|---|
| **HU6.1** | **Unusual Event** |
| Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant **OR** indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the UE emergency classification level.

Plant-Specific

None

**Ginna Basis Reference(s):**

1. NEI 99-01 HU5

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions exist that in the judgment of the Emergency Director warrant declaration of an Alert

**EAL:**

| | |
|---|---|
| **HA6.1** | **Alert** |

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel **OR** damage to site equipment because of hostile action. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE)

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency classification level.

Plant-Specific

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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).

**Ginna Basis Reference(s):**

1. NEI 99-01 HA6

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions existing that in the judgment of the Emergency Director warrant declaration of a Site Area Emergency

**EAL:**

| | |
|---|---|
| **HS6.1** | **Site Area Emergency** |
| Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public **OR** hostile action that results in intentional damage or malicious acts: (1) toward site personnel or equipment that could lead to the likely failure of, or: (2) that prevent effective access to, equipment needed for the protection of the public. **Any** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) beyond the site boundary | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for Site Area Emergency.

Plant-Specific

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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**

The Site Boundary is approximately a 0.3-mile radius around the reactor.

**Ginna Basis Reference(s):**

1. NEI 99-01 HS3

**Category:** H – Hazards and Other Conditions Affecting Plant Safety

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions exist that in the judgment of the Emergency Director warrant declaration of a General Emergency

**EAL:**

| | |
|---|---|
| **HG6.1** | **General Emergency** |
| Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity **OR** hostile action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) offsite for more than the immediate site area | |

**Mode Applicability:**

All

**Basis:**

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for General Emergency.

Plant-Specific

Definitions:

**Hostile Action**

An act toward Ginna 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 Ginna. 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**

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

**Ginna Basis Reference(s):**

1. NEI 99-01 HG2

**Category S – System Malfunction**

EAL Group: Hot Conditions (RCS 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 AC 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 onsite and offsite power sources for the 480V safeguard buses.

2. Loss of 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 the 125 VDC buses.

3. Criticality & RPS Failure

Inadvertent criticalities pose potential personnel safety hazards as well as being indicative of losses of reactivity control.

Events related to failure of the Reactor Protection System (RPS) to initiate and complete automatic 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 automatic 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, RCS and Containment integrity.

4. Inability to Reach or Maintain Shutdown Conditions

System malfunctions may lead to failure of the plant to be brought to the required plant operating condition required by Technical Specifications if a limiting condition for operation (LCO) is not met.

5. Instrumentation

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of annunciators and indicators are in this subcategory.

6. Communications

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

7. Fuel Clad Degradation

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 Category F, Fission Product Barrier Degradation. 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 and/or the Letdown radiation monitor.

8. RCS Leakage

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor 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 RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Containment integrity.

**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** Loss of **all** offsite AC power to 480V safeguards buses for ≥ 15 min.

**EAL:**

| **SU1.1** | **Unusual Event** |
|---|---|
| Loss of **all** offsite AC power, Table S-1, to 480V safeguards buses for ≥ 15 min. (Note 4) | |

Note 4: The ED 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.

| **Table S-1 AC Power Sources** | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

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

**Basis:**

Generic

Prolonged loss of off-site AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power to emergency busses.

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

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., SI pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are not listed in Table S-1 because the availability of power to Buses 17 and 18 alone does not ensure engineered safety features required for hot shutdown and hot standby modes will be operable.

There are three offsite power sources available to these buses (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and an Unusual Event must be declared.

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

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. NEI 99-01 SU1

**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** AC power capability to 480V safeguards buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 480V safeguards bus power

**EAL:**

**SA1.1 Alert**

AC power capability to 480V safeguards buses reduced to a single power source, Table S-1, for ≥ 15 min. (Note 4)

**AND**

**Any** additional single power source failure will result in a complete loss of all 480V safeguards bus power

Note 4: The ED 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.

| Table S-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

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

**Basis:**

<u>Generic</u>

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 480V vital bus AC power . This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all off-site power and loss of on-site emergency generators with only one train of 480V vital busses being backfed from the unit main generator, or the loss of on-site emergency generators with only one train of 480V vital busses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with EAL SS1.1.

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

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., SI pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are not listed in Table S-1 because the availability of power to Buses 17 and 18 alone does not ensure engineered safety features required for hot shutdown and hot standby modes will be operable.

There are three offsite power sources available to these buses (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and an Unusual Event must be declared.

There are two onsite emergency AC power sources available in the hot modes:

- EDG 1A
- EDG 1B

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to be capable of supplying one or more safety-related buses within 15 minutes, an Unusual Event is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to a Site Area Emergency under EAL SS1.1.

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. NEI 99-01 SA5

**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** Loss of **all** offsite and **all** onsite AC power to 480V safeguards buses for ≥ 15 min.

**EAL:**

| |
|---|
| **SS1.1 Site Area Emergency** |
| Loss of **all** offsite and **all** onsite AC power, Table S-1, to 480V safeguards buses for ≥ 15 min. (Note 4) |

Note 4: The ED 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.

| Table S-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

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

**Basis:**

Generic

Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to 480V vital busses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency.

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

Escalation to General Emergency is via EALs in Category F or EAL SG1.1.

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., SI pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are not listed in Table S-1 because the availability of power to Buses 17 and 18 alone does not ensure engineered safety features required for hot shutdown and hot standby modes will be operable.

There are three offsite power sources available to these buses in the cold modes (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

Based on operational experience, if the Unit Auxiliary Transformer backfeed from the Main Transformer is not already aligned, it cannot be considered available/capable of supplying the safeguards buses due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not considered available and Alert must be declared.

There are two onsite emergency AC power sources available in the cold modes:

- EDG 1A
- EDG 1B

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If all sources fail to be capable of supplying all safety-related buses within 15 minutes, a Site Area Emergency is declared under this EAL.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to safeguards buses.

**Ginna Basis Reference(s):**

1. UFSAR Section 8 and Figure 8.1-1 Electrical Distribution System
2. ECA-0.0 Loss of All AC Power
3. NEI 99-01 SS1

**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of AC Power

**Initiating Condition:** Prolonged loss of **all** offsite and **all** onsite AC power to 480V safeguards buses

**EAL:**

**SG1.1 General Emergency**

Loss of **all** offsite and **all** onsite AC power, Table S-1, to 480V safeguards buses

**AND EITHER:**

Restoration of at least one 480V safeguards bus within 4 hours is **not** likely

**OR**

**ORANGE** or **RED** path condition exists F-0.2 Core Cooling

| **Table S-1 AC Power Sources** | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

**Mode Applicability:**

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

**Basis:**

Generic

Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency busses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency.

This EAL is specified to assure that in the unlikely event of a prolonged loss of all AC power to 480V safeguards buses, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one safeguards bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded.

Plant-Specific

Two Class 1E independent trains provide the necessary redundancy on the 480V safeguards system. Train A consists of 480V safeguards buses 14 and 18, while train B consists of safeguards buses 16 and 17. Buses 14 and 16 provide power to engineered safety features that are essential in response to the analyzed events and design basis accidents (e.g., SI pumps, RHR pumps, containment fans, etc.). Buses 17 and 18 provide power to the four service water pumps and are not listed in Table S-1 because the availability of power to Buses 17 and 18 alone does not ensure engineered safety features required for hot shutdown and hot standby modes will be operable.

There are three offsite power sources available to these buses in the cold modes (ref. 1):

- Station Auxiliary Transformer 12A fed from one 34.5 kV transmission line (STA 204 via CKT 7T)
- Station Auxiliary Transformer 12B fed from the 115 kV switchyard (STA 13A via CKT 767) via the 34.5 kV Transformer #6.
- Unit Auxiliary Transformer 11 backfed from the 115 kV switchyard via the 19 kV Main Transformer with the Main Generator bus disconnects (links) removed.

There are two onsite emergency AC power sources available in the cold modes:

- EDG 1A
- EDG 1B

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to safeguards buses.

Ginna is licensed for a four hour Station Black Out (SBO) coping category (ref. 2). The ability of the plant to cope with a four hour SBO duration was based on an assessment of condensate inventory required for decay heat removal, Class 1E battery capacity, compressed air availability or manual operation of certain valves, effects of loss of ventilation, containment isolation valve operability, and reactor coolant inventory loss. A plant-specific analysis indicates that the expected rates of reactor coolant inventory loss under SBO conditions do not result in core uncovery in a SBO for four hours. Therefore, makeup systems in addition to those currently available under SBO conditions are not required to maintain core cooling under natural circulation. Thus, conditions in which restoration of AC power within four hours is not likely are included in the EAL.

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

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

Thus, indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on ED 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 the existence of conditions to Critical Safety Function Status Tree Core Cooling-ORANGE or RED paths (ref. 3).

**Ginna Basis Reference(s):**

1. UFSAR Section 8 Electrical Power and Figure 8.1-1 Electrical Distribution System
2. Ginna Station Blackout Program Section 3.7
3. CSFST for F-0.2 Core Cooling

# 4. NEI 99-01 SG1

**Category:** S – System Malfunction

**Subcategory:** 2 – Loss of DC Power

**Initiating Condition:** Loss of **all** vital DC power for ≥ 15 min.

**EAL:**

| | |
|---|---|
| **SS2.1** | **Site Area Emergency** |
| < 108 VDC on **both** 125 VDC buses 1A and 1B for ≥ 15 min. (Note 4) | |

Note 4: The ED 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.

**Mode Applicability:**

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

**Basis:**

Generic

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

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

Escalation to a General Emergency would occur by EALs in Category R and Category F.

Plant-Specific

The 125 VDC vital system is divided into two independent and isolated channels. Each channel consists of one battery, two battery chargers, one DC bus and one inverter. Each inverter has an associated vital AC distribution panel board. Power to the DC bus, DC unit control panels, and inverters is supplied by the station batteries and/or the battery chargers. Each battery charger is fully rated and can recharge a discharged battery while at the same time supplying the steady state power requirements of the system.

A separate TSC Battery system is designed with an intertie to each of the two main (A and B) distribution panels for use during maintenance and abnormal conditions.

The safety-related station batteries have been sized to carry their expected shutdown loads following a plant trip and loss of offsite power or following a station blackout without battery terminal voltage falling below 108.6 volts for a period of 4 hours (ref. 1).

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

The loss of the TSC Battery does not constitute an entry condition for this EAL.

This EAL is the hot condition equivalent of the cold condition loss of DC power EAL CU2.1.

**Ginna Basis Reference(s):**

1. UFSAR Section 8.3.2 Direct Current Power Systems
2. NEI 99-01 SS3

**Category:** S – System Malfunction

**Subcategory:** 3 – Criticality & RPS Failure

**Initiating Condition:** Inadvertent criticality

**EAL:**

| **SU3.1** | **Unusual Event** |
|---|---|
| An unplanned sustained positive startup rate observed on nuclear instrumentation | |

**Mode Applicability:**

3 - Hot Shutdown, 4 - Hot Standby

**Basis:**

Generic

This EAL addresses inadvertent criticality events. While the primary concern of this EAL is criticality This EAL addresses inadvertent criticality events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

Escalation would be by EALs in Category F, as appropriate to the operating mode at the time of the event.

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the rise in neutron population due to subcritical multiplication. Short-term positive startup rates can also be due to welding activities.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. NEI 99-01 SU8

**Category:** S – System Malfunction

**Subcategory:** 3 – Criticality & RPS Failure

**Initiating Condition:** Automatic trip failed to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

**EAL:**

| | |
|---|---|
| **SA3.1** | **Alert** |

An automatic trip failed to shut down the reactor

**AND**

Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power ≤ 5%

**Mode Applicability:**

1 - Power Operation

**Basis:**

Generic

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

Manual trip actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a trip signal. Thus the plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad barrier or RCS barrier and because of the failure of the Reactor Protection System to automatically shut down the plant.

If manual actions taken at the reactor control console fail to shut down the reactor, the event would escalate to a Site Area Emergency.

Plant-Specific

A reactor trip is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. The symptoms that require an automatic reactor trip are defined in procedure P-1(ref. 4):

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. 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 negative startup rate as nuclear power drops into the source range (ref. 2).

The operator ensures that the reactor has tripped by (ref. 1):

- Verifying that at least one train of reactor trip breakers are open
- Checking that all control rod position rod bottom lights are on
- Observing neutron flux is decreasing

If these responses cannot be verified, operators perform contingency actions that manually insert control rods, open the reactor trip breakers, and tripping the Rod Drive MG sets. Local opening of these breakers requires actions outside of the Control Room; rapid control rod insertion by these methods is therefore not considered a "successful" manual reactor trip. For purposes of emergency classification, a "successful" manual reactor trip, therefore, includes only those immediate actions taken by the reactor operator in the Control Room to actuate reactor trip switches or deenergize 480 V buses 13 and 15 (ref. 1, 2).

In the event that the operator identifies a reactor trip is imminent and successfully initiates a manual reactor trip before the automatic trip setpoint is reached, no declaration is required. 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. If manual reactor trip actions in the Control Room fail to reduce reactor power below 5% (ref. 3), the event escalates to the Site Area Emergency under EAL SS2.1.

**Ginna Basis Reference(s):**

1. E-0 Reactor Trip or Safety Injection
2. FR-S.1 Response to Reactor Restart/ATWS
3. CSFST for F-0.1 Subcriticality
4. P-1 Reactor Control and Protection System
5. NEI 99-01 SA2

**Category:** S – System Malfunction

**Subcategory:** 3 – Criticality & RPS Failure

**Initiating Condition:** Automatic trip and manual actions taken from the reactor control console failed to shut down the reactor

**EAL:**

**SS3.1 Site Area Emergency**

An automatic trip failed to shut down the reactor as indicated by reactor power > 5%

**AND**

Manual actions taken at the reactor control console failed to shut down the reactor as indicated by reactor power > 5%

**Mode Applicability:**

1 - Power Operation

**Basis:**

Generic

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS.

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

Manual scram (trip) actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual trip actions are not considered successful if action away from the reactor control console is required to trip the reactor. This EAL is still applicable even if actions taken away from the reactor control console are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant.

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.

Plant-Specific

A reactor trip is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. The symptoms that require an automatic reactor trip are defined in procedure P-1 (ref. 4): Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. 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 negative startup rate as nuclear power drops into the source range (ref. 2).

The operator ensures that the reactor has tripped by (ref. 1):

- Verifying that at least one train of reactor trip breakers are open
- Checking that all control rod position rod bottom lights are on
- Observing neutron flux is decreasing

If these responses cannot be verified, operators perform contingency actions that manually insert control rods, open the reactor trip breakers, and tripping the Rod Drive MG sets. Local opening of these breakers requires actions outside of the Control Room; rapid control rod insertion by these methods is therefore not considered a "successful" manual reactor trip. For purposes of emergency classification, a "successful" manual reactor trip, therefore, includes only those immediate actions taken by the reactor operator in the Control Room to actuate reactor trip switches or deenergize 480V buses 13 and 15 (ref. 1, 2).

If reactor power is above 5%, the reactor is producing more heat than the maximum decay heat load safety systems are designed to remove (ref. 3). Emergency boration is thus required and there is an actual major failure of a system intended for protection of the public. The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Clad and RCS barriers and warrants declaration of a Site Area Emergency.

**Ginna Basis Reference(s):**

1. E-0 Reactor Trip or Safety Injection
2. FR-S.1 Response to Reactor Restart/ATWS
3. CSFST for F-0.1 Subcriticality
4. Procedure P-1 Reactor Control and Protection System
5. NEI 99-01 SS2

**Category:** S – System Malfunction

**Subcategory:** 3 – Criticality & RPS Failure

**Initiating Condition:** Automatic trip and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists

**EAL:**

| | |
|---|---|
| **SG3.1** | **General Emergency** |

An automatic trip failed to shut down the reactor as indicated by reactor power > 5%

**AND**

**All** manual actions fail to shut down the reactor as indicated by reactor power > 5%

**AND EITHER** of the following exist or have occurred:

RED path condition exists F-0.2 Core Cooling

**OR**

RED path condition exists F-0.3 Heat Sink

**Mode Applicability:**

1 - Power Operation

**Basis:**

Generic

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (5% power). In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier table declaration to permit maximum off-site intervention time.

Plant-Specific

A reactor trip is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. The symptoms that require an automatic reactor trip are defined in procedure P-1 (ref. 6).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. 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 negative startup rate as nuclear power drops into the source range (ref. 2).

The operator ensures that the reactor has tripped by (ref. 1):

- Verifying that at least one train of reactor trip breakers are open
- Checking that all control rod position rod bottom lights are on
- Observing neutron flux is decreasing

If these responses cannot be verified, operators perform contingency actions that manually insert control rods, open the reactor trip breakers, and tripping the Rod Drive MG sets. Local opening of these breakers requires actions outside of the Control Room; rapid control rod insertion by these methods is therefore not considered a "successful" manual reactor trip. For purposes of emergency classification, a "successful" manual reactor trip, therefore, includes only those immediate actions taken by the reactor operator in the Control Room to actuate reactor trip switches or deenergize 480V buses 13 and 15 (ref. 1, 2).

If reactor power is above 5%, the reactor is producing more heat than the maximum decay heat load safety systems are designed to remove (ref. 3). Emergency boration is thus required and there is an actual major failure of a system intended for protection of the public. The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Clad and RCS barriers.

CSFST Core Cooling RED path condition represents a severe challenge to the core cooling function (ref. 4). Core Exit Thermocouples (CETs) are an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. RCS temperatures > 1200 °F or > 700 °F with reactor vessel water level below the top of active fuel signals the transition from a subcooled to a superheated regime. In a superheated regime, heat transfer mechanics are not as efficient as the subcooled condition and could lead to film boiling and a rapid rise in clad temperatures. This condition is considered a Fuel Clad barrier loss condition because the possible rapid rise in clad temperatures may lead to clad failure.

CSFST Heat Sink RED path condition represents a severe challenge to the heat removal function (ref. 5). Inability to remove heat from the RCS to the ultimate heat sink (lake or atmosphere) is a loss of function required for hot shutdown with the reactor at pressure and temperature and thus represents potential loss of the Fuel Clad and RCS barriers. Heat Sink RED path conditions are based on a combination of inadequate S/G level (< 5%) and inadequate feedwater flow (< 200 gpm total S/G feedwater flow).

The combination of these conditions (reactor power greater than 5% with loss of core cooling or inability to remove heat from the RCS) indicates the ultimate heat sink function is under extreme challenge, a core melt sequence may exist and rapid degradation of the fuel clad could begin. To permit maximum offsite intervention time, the General Emergency declaration is appropriate in anticipation of an inevitable General Emergency declaration due to loss and potential loss of fission product barriers.

**Ginna Basis Reference(s):**

1. E-0 Reactor Trip or Safety Injection
2. FR-S.1 Response to Reactor Restart/ATWS
3. CSFST for F-0.1 Subcriticality
4. CSFST for F-0.2 Core Cooling
5. CSFST for F-0.3 Heat Sink
6. P-1 Reactor Control and Protection System
7. NEI 99-01 SG2

**Category:** S – System Malfunction

**Subcategory:** 4 – Inability to Reach or Maintain Shutdown Conditions

**Initiating Condition:** Inability to reach required shutdown within Technical Specification limits

**EAL:**

| | |
|---|---|
| **SU4.1** | **Unusual Event** |
| Plant is **not** brought to required operating mode within Technical Specifications LCO required action completion time | |

**Mode Applicability:**

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

**Basis:**

Generic

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a four hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable required action completion time in the Technical Specifications. An immediate UE is required when the plant is not brought to the required operating mode within the allowable required action completion time in the Technical Specifications. Declaration of a UE is based on the time at which the LCO-specified required action completion time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

Plant-Specific

None

**Ginna Basis Reference(s):**

1. Technical Specifications 3.0, Limiting Conditions for Operations (LCO) Applicability
2. NEI 99-01 SU2

**Category:** S – System Malfunction

**Subcategory:** 5 – Instrumentation

**Initiating Condition:** Unplanned loss of safety system annunciation or indication in the Control Room for ≥ 15 min.

**EAL:**

**SU5.1 Unusual Event**

Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)

Note 4: The ED 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

| Table S-2 Vital Control Room Panels | | | | | | | |
|---|---|---|---|---|---|---|---|
| A | AA | B | C | D | E | F | G |

**Mode Applicability:**

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

**Basis:**

Generic

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical

Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU4.1.

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

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

This UE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a significant transient is in progress during the loss of annunciation or indication.

Plant-Specific

Control Room Panels A through G, Table S-2, provide safety-related indications and annunciation in the Main Control Room (ref. 1, 2).

A 75% loss of annunciators is defined as loss of 6 of 8 annunciator panels listed on Table S-2. Loss of 75% of MCB indications is loss of 75% of the indications on the center and left sections of the main control board indications.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. UFSAR Sections 7.5 Safety-Related Display Instrumentation
2. ER-INST.2 Loss of Annunciators
3. NEI 99-01 SU3

**Category:** S – System Malfunction

**Subcategory:** 5 – Instrumentation

**Initiating Condition:** Unplanned loss of safety system annunciation or indication in the Control Room with either (1) a significant transient in progress, or (2) compensatory indicators are unavailable

**EAL:**

| | |
|---|---|
| **SA5.1** | **Alert** |

Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)

**AND EITHER:**

A significant transient is in progress, Table S-3

**OR**

Compensatory indications are unavailable (PPCS)

Note 4: The ED 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

| Table S-2 Vital Control Room Panels | | | | | | | |
|---|---|---|---|---|---|---|---|
| A | AA | B | C | D | E | F | G |

| Table S-3 Significant Transients |
|---|
| • Automatic turbine runback > 25% thermal power<br>• Electric load rejection > 25% full electrical load<br>• Reactor trip<br>• Safety Injection activation |

**Mode Applicability:**

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

**Basis:**

Generic

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a significant transient.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU4.1.

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

"Compensatory indications" in this context includes computer based information such as Plant Process Computer System.

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

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a significant transient in progress during the loss of annunciation or indication.

<u>Plant-Specific</u>

Control Room Panels A through G, Table S-2, provide safety-related indications and annunciation in the Main Control Room (ref. 1, 2).

PPCS is considered compensatory indication.

Significant transients are listed in Table S-3.

Definitions:

**Unplanned**

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

**Ginna Basis Reference(s):**

1. UFSAR Sections 7.5 Safety-Related Display Instrumentation
2. ER-INST.2 Loss of Annunciators
3. NEI 99-01 SA4

**Category:** S – System Malfunction

**Subcategory:** 5 – Instrumentation

**Initiating Condition:** Inability to monitor a significant transient in progress

**EAL:**

| | |
|---|---|
| **SS5.1** | **Site Area Emergency** |

Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)

**AND**

A significant transient is in progress, Table S-3

**AND**

Compensatory indications are unavailable (PPCS)

Note 4: The ED 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

| **Table S-2 Vital Control Room Panels** | | | | | | | |
|---|---|---|---|---|---|---|---|
| A | AA | B | C | D | E | F | G |

**Table S-3 Significant Transients**

- Automatic turbine runback > 25% thermal power
- Electric load rejection > 25% full electrical load
- Reactor trip
- Safety Injection activation

**Mode Applicability:**

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

**Basis:**

Generic

This EAL is intended to recognize the threat to plant safety associated with the complete loss of capability of the control room staff to monitor plant response to a significant transient.

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

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NOUE is based on EAL SU4.1

A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

Annunciators for this EAL are limited to include those identified in the Abnormal Operating Procedures and in the Emergency Operating Procedures, and in other EALs (e. g., area, process, and/or effluent rad monitors, etc.)]

Indications needed to monitor safety functions necessary for protection of the public include control room indications, computer generated indications and dedicated annunciation capability.

"Compensatory indications" in this context includes computer based information such as Plant Process Computer System.

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

<u>Plant-Specific</u>

Control Room Panels A through G, Table S-2, provide safety-related indications and annunciation in the Main Control Room (ref. 1, 2).

PPCS is considered compensatory indication.

Significant transients are listed in Table S-3.

**Ginna Basis Reference(s):**

1. UFSAR Sections 7.5 Safety-Related Display Instrumentation
2. ER-INST.2 Loss of Annunciators
3. NEI 99-01 SS6

**Category:** S – System Malfunction

**Subcategory:** 6 – Communications

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

**EAL:**

| **SU6.1** | **Unusual Event** |
|---|---|

Loss of **all** Table S-4 onsite (internal) communication methods affecting the ability to perform routine operations

**OR**

Loss of **all** Table S-4 offsite (external) communication methods affecting the ability to perform offsite notifications

**Table S-4 Communications Systems**

| **System** | **Onsite** (internal) | **Offsite** (external) |
|---|---|---|
| Commercial phone system | X | X |
| Direct Dial POTS Lines (Blue Phones) | X | X |
| Plant Page Party system | X | |
| Radios/Walkie Talkies | X | |
| FTS 2001 telephone system (ENS, HPN) | | X |
| Control Room Hard Wired Satellite Phone | | X |
| Control Room Emergency Cell Phone | | X |

**Mode Applicability:**

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

**Basis:**

Generic

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities.

The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to off-site locations, etc.) are being used to make communications possible.

Plant-Specific

Onsite/offsite communications systems are listed in Table S-4 (ref. 1, 2).

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

**Ginna Basis Reference(s):**

1. A-56 Communication Systems at Ginna Station
2. ER-COMM.1 Loss of Communications
3. NEI 99-01 SU6

**Category:** S – System Malfunction

**Subcategory:** 7 – Fuel Clad Degradation

**Initiating Condition:** Fuel clad degradation

**EAL:**

| SU7.1 | Unusual Event |
|---|---|
| RCS specific activity > 60 µCi/gm dose equivalent I-131 | |

**Mode Applicability:**

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

**Basis:**

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

This threshold addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits.

Plant-Specific

This EAL addresses reactor coolant samples exceeding Technical Specification 3.4.16 which is applicable in Modes 1, 2 and in Mode 3 with RCS average temperature ($T_{avg}$) ≥ 500 °F. Though the referenced Technical Specification limits are mode dependent, it is appropriate that the EAL be applicable in all hot modes, as it indicates a potential degradation in the level of safety of the plant. The Technical Specification limits accommodate an iodine spike phenomenon that may occur following changes in thermal power and during reactor startup and shutdown. The Technical Specification LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident (ref. 1).

**Ginna Basis Reference(s):**

1. Technical Specification 3.4.16 Reactor Coolant System - RCS Specific Activity
2. NEI 99-01 SU4

**Category:** S – System Malfunction

**Subcategory:** 7 – Fuel Clad Degradation

**Initiating Condition:** Fuel clad degradation

**EAL:**

| **SU7.2** | **Unusual Event** |
|---|---|
| Valid Letdown Monitor (R-9) reading ≥ 4800 mR/hr | |

**Mode Applicability:**

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

**Basis:**

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

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

Plant-Specific

This EAL addresses indication of gross failed fuel that may be in excess of Technical Specification (ref. 1) coolant activity limits.

The Letdown Line Monitor (R-9) gross radiation channel continuously monitors the activity in a sample drawn from the RCS (NaOH tank room) and actuates an alarm in the Control Room if a predetermined activity level is reached. The high alarm setting of 200 mRad/hr ensures timely detection of failed fuel increases greater than 0.1% (ref. 2, 3, 4).

The 4800 mR/hr value for R-9 is based on total RCS activity corresponding to 60 μCi/gm I-131 equivalent and 1% failed fuel (100 / $\bar{E}$). A shielding calculation was performed to obtain this value (ref. 5).

Definitions:

**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.

**Ginna Basis Reference(s):**

1. Technical Specification 3.4.16 Reactor Coolant System - RCS Specific Activity
2. AP-RCS.3 High Reactor Coolant Activity
3. AR-RMS-9 R9 Letdown Line Monitor
4. P-9 Radiation Monitoring System
5. CALC-2011-0019, "R9 Letdown Line Radiation Monitor NEI 99-01 Rev. 5 Evaluation"
6. NEI 99-01 SU4

**Category:** S – System Malfunction

**Subcategory:** 8 – RCS Leakage

**Initiating Condition:** RCS leakage

**EAL:**

| | |
|---|---|
| **SU8.1** | **Unusual Event** |
| Unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 4) | |
| **OR** | |
| Identified leakage > 25 gpm for ≥ 15 min. (Note 4) | |

Note 4: The ED 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.

**Mode Applicability:**

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

**Basis:**

Generic

This EAL is included as a UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated. 15 minutes allows time to evaluate the source and take corrective actions to isolate the leak.

The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this EAL to the Alert level is via EALs in Category F.

Plant-Specific

Technical Specifications Section 3.4.13 RCS Operational Leakage prescribes RCS leakage limits for pressure boundary (none allowed), unidentified (1 gpm) and identified (10 gpm) leakage (ref. 1). AP-RCS.1 Reactor Coolant Leak provide direction for determining RCS leakage for off normal events and for operations troubleshooting (ref. 2).

**Ginna Basis Reference(s):**

1. Technical Specifications 3.4.13, RCS Operational Leakage
2. AP-RCS.1 Reactor Coolant Leak
3. NEI 99-01 SU5

**Category F – Fission Product Barrier Degradation**

EAL Group: Hot Conditions (RCS 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. Reactor Fuel Clad (FC): The Fuel Clad barrier consists of the zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
C. Containment (CNMT): 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.

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:

*Unusual Event:*
***ANY*** *loss or* ***ANY*** *potential loss of Containment*
*Alert:*

***ANY** loss or **ANY** potential loss of either Fuel Clad or RCS*

*<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*

The logic used for Category F EALs reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. UE EALs associated with RCS and Fuel Clad Barriers are addressed under Category S.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" thresholds existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" thresholds existed, the ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.
- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier.

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

**Initiating Condition:** **ANY** loss or **ANY** potential loss of Containment

**EAL:**

| | |
|---|---|
| **FU1.1** | **Unusual Event** |
| **ANY** loss or **ANY** potential loss of Containment (Table F-1) | |

**Mode Applicability:**

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

**Basis:**

Generic

None

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Fuel Clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Containment barrier in combination with the loss or potential loss of either the Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

**Ginna Basis Reference(s):**

1. NEI 99-01 FU1

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

**Initiating Condition:** **ANY** loss or **ANY** potential loss of either Fuel Clad or RCS

**EAL:**

| | |
|---|---|
| **FA1.1** | **Alert** |
| **ANY** loss or **ANY** potential loss of either Fuel Clad or RCS (Table F-1) | |

**Mode Applicability:**

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

**Basis:**

Generic

None

Plant-Specific

Fuel Clad, RCS 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 RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

**Ginna 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 Shutdown, 4 - Hot Standby

**Basis:**

Generic

None

Plant-Specific

Fuel Clad, RCS 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 RCS 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 RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less imminent.

**Ginna 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 the third barrier

**EAL:**

| |
|---|
| **FG1.1 General Emergency** |
| Loss of **ANY** two barriers |
| **AND** |
| Loss or potential loss of the third barrier (Table F-1) |

**Mode Applicability:**

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

**Basis:**

Generic

None

Plant-Specific

Fuel Clad, RCS 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, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

**Ginna Basis Reference(s):**

1. NEI 99-01 FG1

# ATTACHMENT 2

# FISSION PRODUCT BARRIER
# LOSS/POTENTIAL LOSS MATRIX AND BASES

**Introduction**

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that the three barriers occupy 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. CSFSTs

B. Core Exit TCs

C. Inventory

D. Radiation / Coolant Activity

E. Isolation Status

F. Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the category rows and the Loss/Potential Loss columns. The intersection of each category row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word “None” is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A is “FC Loss A.1,” the third Containment barrier Potential Loss is “CNMT P-Loss B.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 row of fission product barrier Loss and Potential Loss thresholds in that category to determine if any threshold has been exceeded. If a threshold has not been exceeded in that category row, the EAL-user proceeds to the next likely category and continues review of the row of thresholds in the new category

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 (i.e., greater than 1.0E+03 R/hr), a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier exist. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS 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 Matrix**

| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Containment Barrier | |
|---|---|---|---|---|---|---|
| **Category** | **Loss** | **Potential Loss** | **Loss** | **Potential Loss** | **Loss** | **Potential Loss** |
| **A** CSFST | 1. RED path condition exists F-0.2 Core Cooling | 1. ORANGE path condition exists F-0.2 Core Cooling<br>2. RED path condition exists F-0.3 Heat Sink and heat sink is required | None | 1. RED path condition exists F-0.4 Integrity<br>2. RED path condition exists F-0.3 Heat Sink and heat sink is required | None | 1. RED path condition exists F-0.5 Containment |
| **B** Core Exit TCs | 2. Core Exit TCs ≥ 1,200°F | 3. Core Exit TCs ≥ 700°F | None | None | None | 2. Core Exit TCs **cannot** be restored < 1,200°F within 15 min.<br>3. Core Exit TCs ≥ 700ºF **AND** RVLIS level **cannot** be restored > 52% [> 55% adverse CNMT] with no RCPs running within 15 min. |
| **C** Inventory | None | 4. RVLIS level ≤ 52% [ ≤ 55% adverse CNMT] **OR** At least one RCP running RVLIS fluid fraction ≤ 66% | 1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< EOP Fig. MIN SUBCOOLING)<br>2. Ruptured S/G results in an ECCS (SI) actuation | 3. RCS leak rate > 50 gpm with letdown isolated | 1. A containment pressure rise followed by a rapid unexplained drop in containment pressure<br>2. Containment pressure or sump level response **not** consistent with LOCA conditions<br>3. Ruptured S/G is also faulted outside of containment<br>4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable prolonged steam release from affected S/G to the environment | 4. Containment pressure ≥ 60 psig and rising<br>5. Containment hydrogen concentration ≥ 4%<br>6. Containment pressure ≥ 28 psig and < two CRFC units and one CS pump operating per design |
| **D** Radiation / Coolant Activity | 3. Containment radiation monitor R-29/R-30 reading > 1.0E+02 R/hr<br>4. Valid Letdown Monitor (R-9) reading ≥ 24,000 mR/hr<br>5. Coolant activity >300 µCi/gm dose equivalent I-131 | None | 3. Containment radiation monitor R-29/R-30 reading > 1.0E+01 R/hr | None | None | 7. Containment radiation monitor R-29/R-30 reading > 1.0E+03 R/hr |
| **E** Isolation Status | None | None | None | None | 5. Failure of **all** valves in **ANY one** line to close **AND** Direct downstream pathway to the environment exists after containment isolation signal | None |

**Table F-1 Fission Product Barrier Matrix**

| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Containment Barrier | |
|---|---|---|---|---|---|---|
| **Category** | **Loss** | **Potential Loss** | **Loss** | **Potential Loss** | **Loss** | **Potential Loss** |
| **F** **Judgment** | 6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier | 5. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier | 4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier | 4. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier | 6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the containment barrier | 8. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the containment barrier |

**Barrier:** Fuel Clad

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Loss

**Threshold:**

1. RED path condition exists F-0.2 Core Cooling

**Basis:**

Generic

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

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path is given in F-0.2 and indicates significant core exit superheating and core uncovery (ref. 1).

RED path conditions exist if either:

- Core Exit TCs are ≥ 1200°F
- Core Exit TCs are ≥ 700°F with RVLIS ≤ 52% [≤ 55% adverse CNMT] with no RCPs running

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 2):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling

**Barrier:** Fuel Clad

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

1. ORANGE path condition exists F-0.2 Core Cooling

**Basis:**

Generic

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur.

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path is given in F-0.2 and indicates subcooling has been lost and that some fuel clad damage may potentially occur (ref. 1).

ORANGE path Core Cooling conditions exist if, with RCS subcooling < requirements of EOP Fig. MIN SUBCOOLING, either:

- with no RCPs running Core Exit TCs are ≥ 700°F or RVLIS level ≤ 52% [55% adverse CNMT]

  OR

- with at least one RCP running RVLIS fluid fraction ≤ 66%

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 2):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. FR-C.2 Response to Degraded Core Cooling

**Barrier:** Fuel Clad

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 2. RED path condition exists F-0.3 Heat Sink and heat sink is required |

**Basis:**

Generic

Heat Sink - RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.

Plant-Specific

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path in F-0.3 (ref. 1). The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. Procedure FR-H.1, Response to Loss of Secondary Heat Sink, indicates heat sink is required when RCS pressure is greater than any non-faulted SG pressure and RCS cold leg temperature is greater than 350°F (ref. 2).

RED path Heat Sink conditions exist if both of the following:

- Narrow Range level in both S/Gs is ≤ 7% [≤ 25% adverse CNMT]

  AND

- Total feedwater flow to SGs is ≤ 200 gpm

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 3):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

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

**Ginna Basis Reference(s):**

1. CSFST for F-0.3 Heat Sink
2. FR-H.1 Response to Loss of Secondary Heat Sink
2. FR-C.2 Response to Degraded Core Cooling

**Barrier:** Fuel Clad

**Category:** B. Core Exit TCs

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 2. Core Exit TCs ≥ 1,200°F |

**Basis:**

Generic

The 1,200°F reading corresponds to significant superheating of the coolant.

Plant-Specific

Core Exit Thermocouples (TCs) reading at or in excess of 1200ºF corresponds to the CSFST Core Cooling RED path entry condition (ref. 1). Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation (ref. 2). Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery. Events that result in Core Exit TC readings above the loss threshold are classified severe accidents and lead to entry into Severe Accident Management Guidelines (ref. 3).

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. EPIP-2-16 Core Damage Estimation
3. Ginna Severe Accident Management Guidelines
4. FR-C.1 Response To Inadequate Core Cooling

**Barrier:** Fuel Clad

**Category:** B. Core Exit TCs

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 3. Core Exit TCs ≥ 700°F |

**Basis:**

Generic

Core Exit TC readings ≥ 700°F correspond to loss of subcooling.

Plant-Specific

Core Exit Thermocouples (TCs) reading at or in excess of 700°F corresponds to the CSFST Core Cooling ORANGE path entry criteria (ref. 1). Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation (ref. 2). RCS superheat, as indicated by Core Exit TCs reading at or in excess of 700°F, signals the transition from a subcooled to a superheated regime. In a superheated regime, heat transfer mechanics are not as efficient as the subcooled condition and could lead to a rapid rise in clad temperatures. Valid indication of superheat is a potential Fuel Clad barrier loss condition because the possible rapid rise in clad temperatures may lead to clad failure.

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. EPIP-2-16 Core Damage Estimation
3. FR-C.1 Response To Inadequate Core Cooling

**Barrier:** Fuel Clad

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

4. RVLIS ≤ 52% [≤ 55% adverse CNMT]

   **OR**

   At least one RCP running RVLIS fluid fraction ≤ 66%

**Basis:**

Generic

There is no Loss threshold associated with this item.

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

Plant-Specific

The Reactor Vessel water level threshold is used in the EOPs to signal core uncovery and is, therefore, indication of inadequate coolant inventory. If the RVLIS indication drops to 52% [≤ 55% adverse CNMT] **OR** with at least one RCP running RVLIS fluid fraction ≤ 66%, a core covered condition cannot be confirmed. According to the Core Cooling-ORANGE path, this water level indicates subcooling has been lost and that some fuel clad damage may occur. (ref. 1, 2)

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 2):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. FR-C.2 Response to Degraded Core Cooling

**Barrier:** Fuel Clad

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 3. Containment radiation monitor R-29/R-30 reading > 1.0E+02 R/hr |

**Basis:**

Generic

The 1.0E+02 R/hr containment radiation monitor reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment.

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage.

This value is higher than that specified for RCS barrier Loss threshold #3. Thus, this threshold indicates a loss of both the Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with this item.

Plant-Specific

Containment radiation is indicated on R-29 and R-30 (ref. 1).

R-29 & R-30 alert alarms at 1.0E+01 R/hr, indicative of a significant RCS breach (LOCA) in containment (~0.1% gap activity). The R-29 & R-30 high alarm setpoint is set at 1.0E+02 R/hr and is indicative of a significant gap activity release into containment and thus considered a loss of the fuel clad barrier. A reading on containment radiation monitors greater than 1.0E+03 R/hr is indicative of significant fuel activity and thus considered a potential loss of Containment (ref. 2).

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. EPIP-2-16 Core Damage Estimation

**Barrier:** Fuel Clad

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 4. Valid Letdown Line Monitor (R-9) reading ≥ 24,000 mR/hr |

**Basis:**

Generic

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

There is no Potential Loss threshold associated with this item.

Plant-Specific

The Letdown Line Monitor (R-9) gross radiation channel continuously monitors the activity in a sample drawn from the RCS (NaOH tank room) and actuates an alarm in the Control Room if a predetermined activity level is reached. A Letdown Line Monitor reading of 24,000 mR/hr represents fuel clad damage of approximately 5% corresponding to the reactor coolant activity fuel Clad loss threshold of 300 μCi/gm dose equivalent I-131 (ref. 2).

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. CALC-2011-0019, R9 Letdown Line Radiation Monitor NEI 99-01 Rev. 5 Evaluation.

**Barrier:** Fuel Clad

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 5. Coolant activity >300 μCi/gm dose equivalent I-131 |

**Basis:**

Generic

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

There is no Potential Loss threshold associated with this item.

Plant-Specific

None

**Ginna Basis Reference(s):**

1. NEI 99-01 Revision 5, pg 35

**Barrier:** Fuel Clad

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** E. Isolation Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** E. Isolation Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier |

**Basis:**

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

Plant-Specific

The Emergency Director 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.

**Ginna Basis Reference(s):**

None

**Barrier:** Fuel Clad

**Category:** F. Judgment

**Degradation Threat:** Potential Loss

**Threshold:**

5. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier

**Basis:**

Generic

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. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Coordinator judgment that the barrier may be considered potentially lost.

Plant-Specific

The Emergency Director 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.

**Ginna Basis Reference(s):**

None

**Barrier:** Reactor Coolant System

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

1. **RED** path condition exists F-0.4 Integrity

**Basis:**

Generic

RCS Integrity - RED indicates an extreme challenge to the safety function derived from appropriate instrument readings.

There is no Loss threshold associated with this item.

Plant-Specific

Critical Safety Function Status Tree (CSFST) Integrity-RED path is given in F-0.4 and entry is indicative of a direct threat to RCS integrity due to imminent pressurized thermal shock (ref. 1, 2).

RED path Integrity conditions exist if:

- temperature lowers in either RCS cold leg $\geq$ 100°F/hr

  AND

- temperature in either RCS cold leg is $\leq$ 284°F

**Ginna Basis Reference(s):**

1. FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
2. CSFST for F-0.4 Integrity

**Barrier:** Reactor Coolant System

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 2. **RED** path condition exists F-0.3 Heat Sink and heat sink is required |

**Basis:**

Generic

Heat Sink - RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.

There is no Loss threshold associated with this item.

Plant-Specific

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path in F-0.3 (ref. 1). The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. Procedure FR-H.1, Response to Loss of Secondary Heat Sink, indicates heat sink is required when RCS pressure is greater than any non-faulted SG pressure and RCS cold leg temperature is greater than 350°F (ref. 2).

RED path Heat Sink conditions exist if both of the following:

- Narrow Range level in both S/Gs is $\leq$ 7% [$\leq$ 25% adverse CNMT]

  AND

- Total feedwater flow to SGs is $\leq$ 200 gpm

Adverse containment parameters determine when a harsh environment begins to affect instrumentation located inside Containment. The following indications identify that the adverse containment values should be used in the EOPs (ref. 3):

- Containment pressure > 4 psig, or
- Containment radiation > $10^5$ R/hr

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

**Ginna Basis Reference(s):**

1. FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
2. FR-H.1 Response to Loss of Secondary Heat Sink
3. CSFST for F-0.4 Integrity

**Barrier:** Reactor Coolant System

**Category:** B. Core Exit TCs

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** B. Core Exit TCs

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< EOP Fig. MIN SUBCOOLING)

**Basis:**

Generic

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

Plant-Specific

Critical Safety Function Status Trees (CSFST) Core Cooling indicates that if subcooling margin based on core exit TCs is in the Inadequate Subcooling Region of EOP Fig. MIN SUBCOOLING, a loss of RCS subcooling has occurred (ref. 1, 4). E-0, Reactor Trip or Safety Injection and AP-RCS.1, Reactor Coolant Leak, provide appropriate actions to prevent and mitigate the consequences of RCS leakage (ref. 2, 3).

AP-RCS.1 provides a list of conditions that may be observed when excessive RCS leakage occurs and provides appropriate actions to prevent and mitigate the consequences of RCS leakage (ref. 3).

The loss of subcooling as a result of inability to establish RCS heat transfer to the ultimate heat sink is indicative of potential losses of the Fuel Clad and RCS barriers.

**Ginna Basis Reference(s):**

1. F-0.2 CSFST Core Cooling
2. E-0 Reactor Trip or Safety Injection
3. AP-RCS.1 Reactor Coolant Leak
4. EOP Figure MIN SUBCOOLING
5. AP-CVCS.1 CVCS leak

**Barrier:** Reactor Coolant System

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 2. Ruptured S/G results in an ECCS (SI) actuation |

**Basis:**

Generic

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

By itself, this threshold will result in the declaration of an Alert. However, if the SG is also faulted (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment barrier loss thresholds.

There is no potential loss threshold associated with this item.

Plant-Specific

In conjunction with Containment Loss C.3 and the Fuel Clad barrier thresholds, this threshold addresses the full spectrum of Steam Generator Tube Rupture (SGTR) events. A ruptured SG is primary-to-secondary leakage through the steam generator tubes. ECCS (SI) actuation is caused by (ref. 1):

- Containment pressure > 4 psig
- Pressurizer pressure < 1750 psig
- Steam line pressures ≤ 514 psig

Indications of a ruptured S/G include (ref. 2):

- Unexpected rise in either S/G narrow range level
- High radiation on Main Steamline Radiation Monitors
- Local indications of increase steamline radiation

Definitions:

**Ruptured**

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

**Ginna Basis Reference(s):**

1. E-0 Reactor Trip or Safety Injection
2. E-3 Steam Generator Tube Rupture
3. AP-RCS.1 Reactor Coolant Leak

**Barrier:** Reactor Coolant System

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 3. RCS leak rate > 50 gpm with letdown isolated |

**Basis:**

Generic

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

Plant-Specific

The CVCS includes three positive displacement horizontal pumps with a capacity of 46 gpm each (ref. 1). The pressurizer level control program regulates letdown purification subsystem flow by adjusting the letdown flow control valve so that the reactor coolant pump (RCP) controlled leak-off plus the letdown flow matches the input from the operating charging pump. Equilibrium pressurizer level conditions may be disturbed due to RCS temperature changes, power changes, or RCS inventory loss due to leakage. A decrease in pressurizer water level below the programmed level results in a control signal to start one or both standby charging pumps to restore water level. The need for a second or third charging pump to makeup leakage in excess of letdown flow would be indicative of substantial RCS leakage. The single charging pump capacity is rounded up to 50 gpm for this threshold and clearly signals that operation of more than one charging pump is needed (ref. 2).

**Ginna Basis Reference(s):**

1. UFSAR Table 9.3.6 CVCS Performance Parameters
2. UFSAR Section 9.3.4.2.2.2 Charging Pump Control

**Barrier:** Reactor Coolant System

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 3. Containment radiation monitor R-29/R-30 reading > 1.0E+01 R/hr |

**Basis:**

Generic

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

This reading is less than that specified for Fuel Clad barrier threshold 3. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad barrier threshold, fuel damage would also be indicated.

There is no Potential Loss threshold associated with this item.

Plant-Specific

Containment radiation is indicated on R-29 and R-30 (ref. 1).

R-29 & R-30 alert alarms at 1.0E+01 R/hr, indicative of a significant RCS breach (LOCA) in containment (~0.1% gap activity). The R-29 & R-30 high alarm setpoint is set at 1.0E+02 R/hr and is indicative of a significant gap activity release into containment and thus considered a loss of the fuel clad barrier. A reading on containment radiation monitors greater than 1.0E+03 R/hr is indicative of significant fuel activity and thus considered a potential loss of Containment (ref. 2).

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. EPIP-2-16 Core Damage Estimation

**Barrier:** Reactor Coolant System

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** E. Isolation Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** E. Isolation Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

**Basis:**

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

**Ginna Basis Reference(s):**

None

**Barrier:** Reactor Coolant System

**Category:** F. Judgment

**Degradation Threat:** Potential Loss

**Threshold:**

4. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

**Basis:**

Generic

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

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

**Ginna Basis Reference(s):**

None

**Barrier:** Containment

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Containment

**Category:** A. Critical Safety Function Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 1. RED path condition exists F-0.5 Containment |

**Basis:**

Generic

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment.

Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this threshold is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.

There is no Loss threshold associated with this item.

Plant-Specific

Critical Safety Function Status Tree (CSFST) Containment-RED path is given in F-0.5 and is entered if Containment pressure is equal to or greater than 60 psig (ref. 1).

This threshold is indicative of a loss of both RCS and Fuel Clad barriers. This combination of conditions would be expected to require the declaration of a General Emergency.

**Ginna Basis Reference(s):**

1. CSFST for F-0.5 Containment

**Barrier:** Containment

**Category:** B. Core Cooling / Heat Removal

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 2. Core Exit TCs **cannot** be restored < 1,200°F within 15 min. |

**Basis:**

Generic

There is no Loss threshold associated with this item.

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

The function restoration procedures are those emergency operating procedures 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.

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

Plant-Specific

Core Exit Thermocouples (TCs) reading in excess of 1200°F corresponds to the CSFST Core Cooling RED path entry condition (ref. 1). Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation (ref. 2). Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery. Events that result in Core Exit TC readings above the loss threshold are classified severe accidents and lead to entry into Severe Accident Management Guidelines (ref. 3).

Events that result in Core Exit TC readings above the Fuel Clad loss threshold are classified severe accidents and lead to entry into Severe Accident Management Guidelines and signify possible core overheating to the point that clad ballooning/collapse may occur and portions of the core may have melted (ref. 3).

It must also be assumed the loss of RCS inventory is a result of a loss of the RCS barrier. These conditions, if not mitigated, can lead to core melt which in turn may result in a loss of containment. Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The phrase "cannot be restored <" implies Core Exit TC readings have exceeded the threshold temperature and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful (ref. 4). Whether or not guidance is effective should be apparent within fifteen minutes. The ED should make the declaration as soon as it is determined the guidance has not been or will not be effective in restoring temperature below the threshold.

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. EPIP-2-16 Core Damage Estimation
3. Ginna Severe Accident Management Guidelines
4. FR-C.1 Response to Inadequate Core Cooling

**Barrier:** Containment

**Degradation Threat:** Potential Loss

**Category:** B. Core Cooling / Heat Removal

**Threshold:**

| |
|---|
| 3. Core Exit TCs ≥ 700°F<br>**AND**<br>RVLIS level **cannot** be restored > 52% [> 55% adverse CNMT] with no RCPs running within 15 min. |

**Basis:**

Generic

There is no Loss threshold associated with this item.

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

The function restoration procedures are those emergency operating procedures 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.

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

Plant-Specific

Core Exit Thermocouples (TCs) reading in excess of 700°F corresponds to the CSFST Core Cooling ORANGE path entry criteria (ref. 1). Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation (ref. 2). RCS superheat, as indicated by Core Exit TCs reading in excess of 700°F, signals the transition from a subcooled to a superheated regime. In a superheated regime, heat transfer mechanics are not as efficient as the subcooled condition and could lead to a rapid rise in clad temperatures. Valid indication of superheat is a potential Fuel Clad barrier loss condition because the possible rapid rise in clad temperatures may lead to clad failure.

This threshold indicates: subcooling has been lost (Core Exit TC readings ≥ 700°F), the core is uncovered and some fuel clad damage may be occurring (ineffective functional restoration procedures) (ref. 1, 3). It must be assumed that the loss of RCS inventory is a result of a loss of the RCS barrier.

These conditions, if not mitigated, can lead to core melt which in turn may result in a loss of containment. Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The phrase "cannot be restored <" implies Core Exit TC readings have exceeded the threshold temperature and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful (ref. 3). Whether or not guidance is effective should be apparent within fifteen minutes. The ED should make the declaration as soon as it is determined the guidance has not been or will not be effective in restoring temperature below the threshold.

**Ginna Basis Reference(s):**

1. CSFST for F-0.2 Core Cooling
2. EPIP-2-16 Core Damage Estimation
3. FR-C.2 Response to Degraded Core Cooling
4. Drawing 03021-0687 Reactor Vessel Level Monitoring System Elevations

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 1. A containment pressure rise followed by a rapid unexplained drop in containment pressure |

**Basis:**

Generic

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

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

Plant-Specific

UFSAR Figure 15.6-34 describes containment pressure response for a large break LOCA (ref. 1). Containment pressure peaks at approximately 45 psig at approximately 25 seconds after event initiation.

**Ginna Basis Reference(s):**

1. UFSAR Figure 15.6-34 Containment Pressure Used for the R.E. Ginna Best-Estimate Large Break LOCA

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

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

**Basis:**

Generic

Containment sump levels should increase as a result of mass and energy release into containment from a LOCA. Thus, sump level not increasing indicates containment bypass and a loss of containment integrity.

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

Plant-Specific

The containment pressure and temperature response and containment sump water temperature response versus time are given in UFSAR Figures 6.2-1 through 6.2-6 for the most severe LOCAs (ref. 1).

**Ginna Basis Reference(s):**

1. UFSAR Figures 6.2-1 through 6.2-6

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 3. Ruptured S/G is also faulted outside of containment |

**Basis:**

Generic

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

Users should realize that this threshold and containment loss C.4 could be considered redundant. This was recognized during the development process. The inclusion of a threshold that uses Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

This threshold results in a UE for smaller breaks that; (1) do not exceed the normal charging capacity threshold in RCS leak rate barrier Potential Loss threshold, or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss threshold. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this threshold would exist in conjunction with RCS barrier thresholds and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

This threshold addresses the condition in which a ruptured steam generator is also faulted. This condition represents a bypass of the RCS and containment barriers and is a subset of the containment loss C.4. In conjunction with RCS leak rate barrier loss threshold, this would always result in the declaration of a Site Area Emergency.

Plant-Specific

A faulted S/G means the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized (ref. 1). A ruptured S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection (ref. 2).

Definitions:

**Faulted**

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

**Ruptured**

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

**Ginna Basis Reference(s):**

1. E-2 Faulted Steam Generator Isolation
2. E-3 Steam Generator Tube Rupture

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 4. Primary-to-secondary leakrate > 10 gpm<br>**AND**<br>Unisolable prolonged steam release from affected S/G to the environment |

**Basis:**

Generic

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

Users should realize that the this loss threshold and containment loss C.3 could be considered redundant. This was recognized during the development process. The inclusion of an threshold that uses Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

This threshold results in a UE for smaller breaks that; (1) do not exceed the normal charging capacity threshold in RCS leak rate barrier Potential Loss threshold, or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss threshold. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this threshold would exist in conjunction with RCS barrier thresholds and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

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

Plant-Specific

Cooldowns conducted to allow controlled isolation of the affected S/G per emergency procedures are not considered prolonged releases. The criterion for prolonged release is met if the objective of E-3 to isolate the affected S/G cannot be met (ref. 2).

An ARV or Safety valve performing as designed is not considered a "failed" barrier.

Definitions:

**Unisolable**

A breach or leak that cannot be promptly isolated from the Main Control Board.

**Ginna Basis Reference(s):**

1. ECA-1.2 LOCA Outside Containment
2. E-3 Steam Generator Tube Rupture
3. F-0.2 Core Cooling

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 4. Containment pressure ≥ 60 psig and rising |

**Basis:**

Generic

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

Plant-Specific

This threshold is the containment design pressure and is in excess of that expected from the design basis loss of coolant accident (LOCA) (ref. 1, 2). Proper actuation and operation of the containment spray system when required should maintain containment pressure well below the design pressure. The peak containment pressure of 45 psig occurs ~ 25 seconds after event initiation for the most limiting design basis LOCA (ref. 3). The pressure-time responses for the spectrum of LOCAs considered in the plant design basis are described in Section 15 of the UFSAR, Accident Analyses. The threshold is therefore indicative of a loss of both RCS and Fuel Clad barriers in that it should not be reached without severe core degradation (metal-water reaction) or failure to scram in combination with RCS breach. This condition would be expected to require the declaration of a General Emergency.

**Ginna Basis Reference(s):**

1. CSFST for F.0.5 Containment
2. UFSAR 3.1.2.2.7 General Design Criterion 16 – Containment Design
3. UFSAR Figure 15.6-34 Containment Pressure Used for the R.E. Ginna Best-Estimate Large Break LOCA

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 5. Containment hydrogen concentration ≥ 4% |

**Basis:**

Generic

Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to containment potential loss threshold A.1.

Plant-Specific

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 Containment. However, Containment monitoring (CH-EPIP-CVH2) and/or sampling should be performed to verify this assumption (ref. 1). A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4.1% (rounded to 4%) by volume (ref. 2).

After a LOCA, the containment atmosphere is a homogeneous mixture of steam, air, solid and gaseous fission products, hydrogen, and water droplets containing boron and sodium hydroxide. During and following a LOCA, the hydrogen concentration in the Containment results from radiolytic decomposition of water, metal-water reaction, and aluminum/zinc reaction with the spray solution (ref. 2). If hydrogen concentration reaches or exceeds the lower flammability limit (4%) in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Clad and RCS barriers with the potential loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

**Ginna Basis Reference(s):**

1. SACRG-1 Severe Accident Control Room Guideline Initial Response
2. UFSAR 1.5.10 Development of Containment Hydrogen Recombiner

**Barrier:** Containment

**Category:** C. Inventory

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 6. Containment pressure ≥ 28 psig and < two CRFC units and one CS pump operating per design |

**Basis:**

Generic

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

Plant-Specific

Two means of post accident containment heat removal are provided; Containment Spray System and Containment Recirc Fan Cooler (CRFC) units. At least one train of each of these systems is required to provide sufficient steam-condensing capacity to ensure against containment overstress and to remove residual and chemical heat (ref. 1, 2).

The CRFC system is comprised of four CRFC units, two of which are required in the post accident condition (ref. 3, 4). Each containment aircooling unit consists of cooling coils, accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path. Following an SI actuation signal, CRFC System fans are designed to start automatically (ref. 4).

Each of two containment spray trains consists of a spray pump, spray header, nozzles, valves, piping, instruments, and controls to ensure an operable flow path capable of taking suction from the RWST upon an actuation signal (ref. 4).

During a steam line break or LOCA, a minimum of two CRFC units and one Containment Spray (CS) pump are required to maintain peak pressure and temperature below design limits (ref. 4).

The containment hi-hi pressure setpoint (28 psig) is the pressure at which the equipment should actuate and begin performing its function (ref. 5).

**Ginna Basis Reference(s):**

1. UFSAR Section 6.2.2 Containment Heat Removal Systems
2. UFSAR Section 6.2.1.2.3 Secondary System Pipe Break Analysis
3. UFSAR Section 6.2.2.1.3 Design Evaluation
4. Technical Specifications B 3.6. Containment Systems
5. CSFST for F-0.5 Containment

**Barrier:** Containment

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Containment

**Category:** D. Radiation / Coolant Activity

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 7. Containment radiation monitor R-29/R-30 reading > 1.0E+03 R/hr |

**Basis:**

Generic

There is no Loss threshold associated with this item.

The site specific reading is a value which indicates significant fuel damage well in excess of the thresholds associated with both loss of Fuel Clad and loss of RCS barriers. A major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of fuel clad allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted.

Plant-Specific

Containment radiation is indicated on R-29 and R-30 (ref. 1).

R-29 & R-30 alert alarms at 10 R/hr, indicative of a significant RCS breach (LOCA) in containment (~0.1% gap activity). The R-29 & R-30 high alarm setpoint is set at 100 R/hr and is indicative of a significant gap activity release into containment and thus considered a loss of the fuel clad barrier. A reading on containment radiation monitors greater than 1.0E+03 R/hr is indicative of significant fuel activity and thus considered a potential loss of Containment (ref. 2).

The containment radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier. NUREG-1228 "Source Term Estimations During Incident Response to Severe Nuclear Power Plant Accidents" states that such readings do not exist when the amount of clad damage is less than 20% (ref. 3). A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure into the reactor coolant has occurred. Regardless of whether the Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Containment barrier.

The reading is higher than that specified for Fuel Clad barrier Loss #3 and RCS barrier Loss #3. Containment radiation readings at or above the Containment barrier potential loss threshold, therefore, signify a loss of two fission product barriers and potential loss of a third, indicating the need to upgrade the emergency classification to a General Emergency.

**Ginna Basis Reference(s):**

1. P-9 Radiation Monitoring System
2. EPIP-2-16 Core Damage Assessment Estimation
3. NUREG-1228 Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents

**Barrier:** Containment

**Category:** E. Isolation Status

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 5. Failure of **all** valves in **ANY one** line to close<br>**AND**<br>Direct downstream pathway to the environment exists after containment isolation signal |

**Basis:**

Generic

This threshold addresses incomplete containment isolation that allows direct release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no Potential Loss threshold associated with this item.

Plant-Specific

None

**Ginna Basis Reference(s):**

1. EOP Attachment 27 Attachment Automatic Action Verification
2. EOP Attachment 3 Attachment CI/CVI

**Barrier:** Containment

**Category:** E. Isolation Status

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Containment

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier |

**Basis:**

Generic

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

The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

**Ginna Basis Reference(s):**

None

**Barrier:** Containment

**Category:** F. Judgment

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 8. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier |

**Basis:**

<u>Generic</u>

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

The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

<u>Plant-Specific</u>

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

**Ginna Basis Reference(s):**

None

# ATTACHMENT (3)

# EAL COMPARISON MATRIX

**Constellation Energy®**

R.E. Ginna Nuclear Power Plant

# NEI 99-01 Revision 5 to GINNA
# EAL Comparison Matrix

Revision 0

## Table of Contents

## Introduction

This document provides a line-by-line comparison of the Initiating Conditions (ICs), Mode Applicability and Emergency Action Levels (EALs) in NEI 99-01 Rev. 5 Final, Methodology for Development of Emergency Action Levels, February 2008 (ADAMS Accession Number ML080450149),and the R.E. Ginna Nuclear Power Plant (Ginna) ICs, Mode Applicability and EALs. This document provides a means of assessing Ginna differences and deviations from the NRC endorsed guidance given in NEI 99-01. Discussion of Ginna EAL bases and lists of source document references are given in the EAL Technical Bases Document. It is, therefore, advisable to reference the EAL Technical Bases Document for background information while using this document.

## Comparison Matrix Format

The ICs and EALs discussed in this document are grouped according to NEI 99-01 Recognition Categories. Within each Recognition Category, the ICs and EALs are listed in tabular format according to the order in which they are given in NEI 99-01. Generally, each row of the comparison matrix provides the following information:

- NEI EAL/IC identifier
- NEI EAL/IC wording
- Ginna EAL/IC identifier
- Ginna EAL/IC wording
- Description of any differences or deviations

## EAL Wording

In Section 4.2, NEI recommends the following: "The method of [EAL] presentation should be one with which the operations and health physics staff are comfortable. As is the case for emergency procedures, bases for steps should be in a separate (or separable) document suitable for training and for reference by emergency response personnel and offsite agencies. Each nuclear plant should already have presentation and human factors standards as part of its procedure writing guidance. EALs that are consistent with those procedure writing standards (in particular, emergency operating procedures which most closely correspond to the conditions under which EALs must be used) should be the norm for each utility."

To assist the Emergency Director (ED), the Ginna EALs have been written in a clear and concise style (to the extent that the differences from the NEI EAL wording could be reasonably documented and justified). As a result, unnecessary words have been removed from the Ginna EALs to reduce EAL-user reading burden to the extent practicable.

The wording reduction gained from elimination of a few characters in a given EAL may not appear to be advantageous within the context of one EAL. When applied to the composite set of EALs, however, significant gains are realized and reading efficiency is improved. This supports timely and accurate classification in the tense atmosphere of an emergency event. The EAL differences introduced to reduce reading burden comprise almost all of the differences justified in this document.

## EAL Emphasis Techniques

Due to the width of the table columns and table formatting constraints in this document, line breaks and indentation may differ slightly from the appearance of comparable wording in the source documents. NEI 99-01 is the source document for the NEI EALs; the Ginna EAL Technical Bases Document for the Ginna EALs.

Development of the Ginna IC/EAL wording has attempted to minimize inconsistencies and apply sound human factors principles. As a result, differences occur between NEI and Ginna ICs/EALs for these reasons alone. When such difference may imply a technical difference in the associated NEI IC/EAL, the difference is identified and a justification provided.

The print and paragraph formatting conventions summarized below guide presentation of the Ginna EALs in accordance with the EAL writing criteria. Space restrictions in the EAL table of this document sometimes override these criteria in cases when following the criteria would introduce undesirable complications in the EAL layout.

- Upper case print is reserved for system abbreviations, logic terms (and, or, etc. when not used as a conjunction), annunciator window engravings.
- Bold font is used for logic terms, negative terms (**not**, **cannot**, etc.), **ANY**, **all**.

- Underscore is avoided as it can interfere with text in narrow line spacing.
- Three or more items in a list are normally introduced with "**ANY** of the following" or "**all** of the following." Items of the list begin with bullets when a priority or sequence is not implied.
- The use of **AND/OR** logic within the same EAL has been avoided when possible. When such logic cannot be avoided, indentation and separation of subordinate contingent phrases is employed.

**Global Differences**

The differences listed below generally apply throughout the set of EALs and are not repeated in the Justification sections of this document. The global differences do not decrease the effectiveness of the intent of NEI 99-01.

1. The NEI phrase "Notification of Unusual Event" has been changed to "Unusual Event" or abbreviated "UE" to reduce EAL-user reading burden.
2. NEI 99-01 IC Example EALs are implemented in separate plant EALs to improve clarity and readability. For example, NEI lists all IC HU1 Example EALs under one IC. The corresponding Ginna EALs appear as unique EALs (e.g., HU1.1 through HU1.5).
3. Mode applicability identifiers (numbers/letter) modify the NEI 99-01 mode applicability names as follows: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby, 5 - Cold Shutdown, 6 - Refuel, D – Defueled. NEI 99-01 defines Defueled as follows: "Reactor Vessel contains no irradiated fuel (full core off-load during refueling or extended outage)."
4. NEI 99-01 uses the terms greater than, less than, greater than or equal to, for [x] minutes or longer, etc. in the wording of some ICs and example EALs. For consistency and to reduce EAL-user reading burden, Ginna has adopted use of boolean symbols (>, <, ≥, ≤) in place of the NEI 99-01 text modifiers.
5. "min." is the standard abbreviation for "minutes" and is used to reduce EAL-user reading burden.
6. IC/EAL identification:

- NEI Recognition Category A "Abnormal Radiation Levels/ Radiological Effluents" has been changed to Category R "Abnormal Rad Release / Rad Effluent." The designator "R" is more intuitively associated with radiation (rad) or radiological events. NEI IC designators beginning with "A" have likewise been changed to "R."
- NEI 99-01 defines the thresholds requiring emergency classification (example EALs) and assigns them to ICs which, in turn, are grouped in "Recognition Categories." The Recognition Categories, however, are so broad and the IC descriptions are so varied that an EAL is difficult to locate in a timely manner when the EAL-user must refer to a set of EALs with the NEI organization and identification scheme. The NEI document clearly states that the EAL/IC/Recognition Category scheme is **not** intended to be the plant-specific EAL scheme for any plant, and appropriate human factors principles should be applied to development of an EAL scheme that helps the EAL-user make timely and accurate classifications. Ginna endeavors to improve upon the NEI EAL organization and identification scheme to enhance usability of the plant-specific EAL set. To this end, the Ginna IC/EAL scheme includes the following features:
    a. Division of the NEI EAL set into three 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, Refuel 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.

b. Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The Ginna EAL categories/subcategories and their relationship to NEI Recognition Categories are listed in Table 1.

c. Unique identification of each EAL – Four characters comprise the EAL identifier as illustrated in Figure 1.

**Figure 1 – EAL Identifier**

**EAL Identifier**

**XXX.X**

Category (R, H, E, S, F, C)
Emergency classification (G, S, A, U)
Sequential number within subcategory/classification
Subcategory number (1 if no subcategory)

The first character is a letter associated with the category in which the EAL is located. The second character is a letter associated with the emergency classification level. (G for General Emergency, S for Site Area Emergency, A for Alert, and U for Notification of Unusual Event). The third character is a number associated with one or more subcategories within a given category. Subcategories are sequentially numbered beginning with the number "1." If a category does not have a subcategory, this character is assigned the number "1." The fourth character is a number preceded by a period for each EAL within a subcategory. EALs are sequentially numbered within the emergency classification level of a subcategory beginning with the number "1."

The EAL identifier is designed to fulfill the following objectives:

- Uniqueness – The EAL identifier ensures that there can be no confusion over which EAL is driving the need for emergency classification.
- Speed in locating the EAL of concern – When the EALs are displayed in a matrix format, knowledge of the EAL identifier alone can lead the EAL-user to the location of the EAL within the classification matrix. The identifier conveys the category, subcategory and classification level. This assists ERO responders (who may not be in the same facility as the ED) to find the EAL of concern in a timely manner without the need for a word description of the classification threshold.
- Possible classification upgrade – The category/subcategory/identifier scheme helps the EAL-user find higher emergency classification EALs that may become active if plant conditions worsen.

Note that the NEI 99-01 identifier only identifies the IC, not the specific example EAL threshold. The NEI scheme, therefore, does not fulfill the above objectives which are desirable in facilitating timely and accurate emergency classification.

Table 2 lists the Ginna ICs and EALs that correspond to the NEI ICs/Example EALs when the above EAL/IC organization and identification scheme is implemented.

**Differences and Deviations**

According to NRC Regulatory Issue Summary (RIS) 2003-18 "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels" Supplements 1 and 2, a difference is an EAL change in which the basis scheme guidance differs in wording but agrees in meaning and intent, such that classification of an event would be the same, whether using the basis scheme guidance or the Ginna EAL. A deviation is an EAL change in which the basis scheme guidance differs in wording and is altered in meaning or intent, such that classification of the event could be different between the basis scheme guidance and the Ginna proposed EAL.

Administrative changes that do not actually change the textual content are neither differences nor deviations. Likewise, any format change that does not alter the wording of the IC or EAL is considered neither a difference nor a deviation.

The following are examples of differences:

- Choosing the applicable EAL based upon plant type (i.e., BWR vs. PWR).
- Using a numbering scheme other than that provided in NEI 99-01 that does not change the intent of the overall scheme.
- Where the NEI 99-01 guidance specifically provides an option to not include an EAL if equipment for the EAL does not exist at Ginna (e.g., automatic real-time dose assessment capability).
- Pulling information from the bases section up to the actual EAL that does not change the intent of the EAL.
- Choosing to state ALL Operating Modes are applicable instead of stating N/A, or listing each mode individually under the Abnormal Rad Level/Radiological Effluent and Hazard and Other Conditions Affecting Plant Safety sections.
- Using synonymous wording (e.g., greater than or equal to vs. at or above, less than or equal to vs. at or below, greater than or less than vs. above or below, etc.)
- Adding Ginna equipment/instrument identification and/or noun names to EALs.
- Combining like ICs that are exactly the same but have different operating modes as long as the intent of each IC is maintained and the overall progression of the EAL scheme is not affected.
- Any change to the IC and/or EAL, and/or basis wording, as stated in NEI 99-01, that does not alter the intent of the IC and/or EAL, i.e., the IC and/or EAL continues to:
    - Classify at the correct classification level.
    - Logically integrate with other EALs in the EAL scheme.
    - Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01 definitions as the intent is for all NEI 99-01 users to have a standard set of defined terms as defined in NEI 99-01. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording in NEI 99-01 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 is encouraged since the intent is for all users to have a standard set of defined terms as defined in NEI 99-01.
- Any change to the IC and/or EAL, and/or basis wording as stated in NEI 99-01 that does alter the intent of the IC and/or EAL, i.e., the IC and/or EAL:
    - Does not classify at the classification level consistent with NEI 99-01.
    - Is not logically integrated with other EALs in the EAL scheme.
    - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The "Difference/Deviation Justification" columns in the remaining sections of this document identify each difference between the NEI 99-01 IC/EAL wording and the Ginna IC/EAL wording. An explanation that justifies the reason for each difference is then provided. If the difference is determined to be a deviation, a statement is made to that effect and explanation is given that states why classification may be different from the NEI 99-01 IC/EAL and the reason for its acceptability. In all cases, however, the differences and deviations do not decrease the effectiveness of the intent of NEI 99-01. A summary list of Ginna EAL deviations from NEI 99-01 is given in Table 3.

## Table 1 – Ginna EAL Categories/Subcategories

| Ginna EALs | | NEI Recognition Category |
|---|---|---|
| **Category** | **Subcategory** | |
| Group: Any Operating Mode: | | |
| R – Abnormal Rad Release/Rad Effluent | 1 – Offsite Rad Conditions<br>2 – Onsite Rad Conditions & Spent Fuel Events<br>3 – CR/CAS Rad | Abnormal **R**ad Levels/Radiological Effluent EALs |
| H – Hazards and Other Conditions Affecting Plant Safety | 1 – Natural or Destructive Phenomena<br>2 – Fire or Explosion<br>3 – Hazardous Gas<br>4 – Security<br>5 – Control Room Evacuation<br>6 – Judgment | **H**azards and Other Conditions Affecting Plant Safety EALs |
| E - ISFSI | None | ISFSI EALs |
| Group: Hot Conditions: | | |
| S – System Malfunction | 1 – Loss of AC Power<br>2 – Loss of DC Power<br>3 – Criticality & RPS Failure<br>4 – Inability to Reach or Maintain Shutdown Conditions<br>5 – Instrumentation<br>6 – Communications<br>7 – Fuel Clad Degradation<br>8 – RCS Leakage | **S**ystem Malfunction EALs |
| F – Fission Product Barrier | None | **F**ission Product Barrier EALs |
| Group: Cold Conditions: | | |
| C – Cold Shutdown/Refuel System Malfunction | 1 – Loss of AC Power<br>2 – Loss of DC Power<br>3 – RCS Level<br>4 – RCS Temperature<br>5 – Communications<br>6 – Inadvertent Criticality | **C**old Shutdown / Refueling System Malfunction EALs |

## Table 2 – NEI / Ginna EAL Identification Cross-Reference

| NEI | | Ginna | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| AU1 | 1 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RU1.1 |
| AU1 | 2 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RU1.1 |
| AU1 | 3 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RU1.2 |
| AU1 | 4 | N/A | N/A |
| AU1 | 5 | N/A | N/A |
| AU2 | 1 | R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events | RU2.1 |
| AU2 | 2 | R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events | RU2.2 |
| AA1 | 1 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RA1.1 |
| AA1 | 2 | N/A | N/A |
| AA1 | 3 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RA1.2 |
| AA1 | 4 | N/A | N/A |
| AA1 | 5 | N/A | N/A |
| AA2 | 1 | R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events | RA2.2 |
| AA2 | 2 | R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events | RA2.1 |

| NEI | | Ginna | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| AA3 | 1 | R – Abnormal Rad Release / Rad Effluent, 3 – CR/CAS Radiation | RA3.1 |
| AS1 | 1 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RS1.1 |
| AS1 | 2 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RS1.2 |
| AS1 | 3 | N/A | N/A |
| AS1 | 4 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RS1.3 |
| AG1 | 1 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RG1.1 |
| AG1 | 2 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RG1.2 |
| AG1 | 3 | N/A | N/A |
| AG1 | 4 | R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | RG1.3 |
| CU1 | 1, 2 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CU3.1 |
| CU2 | 1 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CU3.2 |
| CU2 | 2 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CU3.3 |
| CU3 | 1 | C – Cold SD/ Refuel System Malfunction, 1 – Loss of AC Power | CU1.1 |
| CU4 | 1 | C – Cold SD/ Refuel System Malfunction, 4 – RCS Temperature | CU4.1 |
| CU4 | 2 | C – Cold SD/ Refuel System Malfunction, 4 – RCS Temperature | CU4.2 |
| CU6 | 1, 2 | C – Cold SD/ Refuel System Malfunction, 5 – Communications | CU5.1 |
| CU7 | 1 | C – Cold SD/ Refuel System Malfunction, 2 – Loss of DC Power | CU2.1 |
| CU8 | 1 | N/A | N/A |

| NEI | | Ginna | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| CU8 | 2 | C – Cold SD/ Refuel System Malfunction, 6 – Inadvertent Criticality | CU6.1 |
| CA1 | 1, 2 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CA3.1 |
| CA3 | 1 | C – Cold SD/ Refuel System Malfunction, 1 – Loss of AC Power | CA1.1 |
| CA4 | 1, 2 | C – Cold SD/ Refuel System Malfunction, 4 – RCS Temperature | CA4.1 |
| CS1 | 1 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CS3.1 |
| CS1 | 2 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CS3.2 |
| CS1 | 3 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CS3.3 |
| CG1 | 1 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CG3.1 |
| CG1 | 2 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Level | CG3.2 |
| D-AU1<br>D-AU2<br>D-SU1<br>D-HU1<br>D-HU2<br>D-HU3<br>D-AA1<br>D-AA2<br>D-HA1<br>D-HA2 | | N/A | N/A |
| E-HU1 | 1 | N/A | N/A |
| FU1 | 1 | F – Fission Product Barriers | FU1.1 |

| NEI | | Ginna | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| FA1 | 1 | F – Fission Product Barriers | FA1.1 |
| FS1 | 1 | F – Fission Product Barriers | FS1.1 |
| FG1 | 1 | F – Fission Product Barriers | FG1.1 |
| HU1 | 1 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.1 |
| HU1 | 2 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.2 |
| HU1 | 3 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.3 |
| HU1 | 4 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.4 |
| HU1 | 5 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.5 |
| HU2 | 1 | H – Hazards, 2 – Fire or Explosion | HU2.1 |
| HU2 | 2 | H – Hazards, 2 – Fire or Explosion | HU2.2 |
| HU3 | 1 | H – Hazards, 3 – Hazardous Gas | HU3.1 |
| HU3 | 2 | H – Hazards, 3 – Hazardous Gas | HU3.2 |
| HU4 | 1, 2, 3 | H – Hazards, 4 – Security | HU4.1 |
| HU5 | 1 | H – Hazards, 6 – Judgment | HU6.1 |
| HA1 | 1 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.1 |
| HA1 | 2 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.2 |
| HA1 | 3 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.3 |
| HA1 | 4 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.4 |

| NEI | | Ginna | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| HA1 | 5 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.6 |
| HA1 | 6 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.5 |
| HA2 | 1 | H – Hazards, 2 – Fire or Explosion | HA2.1 |
| HA3 | 1 | H – Hazards, 3 – Hazardous Gas | HA3.1 |
| HA4 | 1, 2 | H – Hazards, 4 – Security | HA4.1 |
| HA5 | 1 | H – Hazards, 5 – Control Room Evacuation | HA5.1 |
| HA6 | 1 | H – Hazards, 6 –Judgment | HA6.1 |
| HS2 | 1 | H – Hazards, 5 – Control Room Evacuation | HS5.1 |
| HS3 | 1 | H – Hazards, 6 – Judgment | HS6.1 |
| HS4 | 1 | H – Hazards, 4 – Security | HS4.1 |
| HG1 | 1 | H – Hazards, 4 – Security | HG4.1 |
| HG1 | 2 | H – Hazards, 4 – Security | HG4.2 |
| HG2 | 1 | H – Hazards, 6 – Judgment | HG6.1 |
| SU1 | 1 | S – System Malfunction, 1 – Loss of AC Power | SU1.1 |
| SU2 | 1 | S – System Malfunction, 4 – Inability to Reach or Maintain Shutdown Conditions | SU4.1 |
| SU3 | 1 | S – System Malfunction, 5 – Instrumentation / Communications | SU5.1 |
| SU4 | 1 | S – System Malfunction, 7 – Fuel Clad Degradation | SU7.2 |
| SU4 | 2 | S – System Malfunction, 7 – Fuel Clad Degradation | SU7.1 |

| NEI | | Ginna | |
|---|---|---|---|
| **IC** | **Example EAL** | **Category and Subcategory** | **EAL** |
| SU5 | 1, 2 | S – System Malfunction, 8 – RCS Leakage | SU8.1 |
| SU6 | 1, 2 | S – System Malfunction, 6 – Instrumentation / Communications | SU6.1 |
| SU8 | 1 | N/A | N/A |
| SU8 | 2 | S – System Malfunction, 3 – Criticality & RPS Failure | SU3.1 |
| SA2 | 1 | S – System Malfunction, 2 – Criticality & RPS Failure | SA3.1 |
| SA4 | 1 | S – System Malfunction, 5 – Instrumentation / Communications | SA5.1 |
| SA5 | 1 | S – System Malfunction, 1 – Loss of AC Power | SA1.1 |
| SS1 | 1 | S – System Malfunction, 1 – Loss of AC Power | SS1.1 |
| SS2 | 1 | S – System Malfunction, 3 – Criticality & RPS Failure | SS3.1 |
| SS3 | 1 | S – System Malfunction, 1 – Loss of DC Power | SS2.1 |
| SS6 | 1 | S – System Malfunction, 5 – Instrumentation / Communications | SS5.1 |
| SG1 | 1 | S – System Malfunction, 1 – Loss of AC Power | SG1.1 |
| SG2 | 1 | S – System Malfunction, 3 – Criticality & RPS Failure | SG3.1 |

## Table 3 – Summary of Deviations

| NEI | | Ginna EAL/FPB Threshold | Description |
|---|---|---|---|
| IC | Example EAL | | |
| SU5 | 1, 2 | SU8.1 | The phrase "for ≥ 15 min. (Note 4)" has been added to the Ginna EAL to allow mitigation by operating procedures prior to declaration.<br>This deviation is based on the EAL FAQ #34 dated September 2010 (ML102030330). The current EAL does not provide a threshold time to evaluate or mitigate the event such as the EAL for loss of off-site power or the fire EAL. Providing time to isolate a leak using plant procedures would eliminate unnecessary classifications and notifications to off-site organizations. Off-site response organizations have provided feedback during the EAL upgrade process to eliminate unnecessary classifications that can be easily addressed by plant staff. There should be time for the Control Room Operators to use procedures to attempt identification and isolate of the leakage prior to classification. The EAL would then be based upon the inability to maintain RCS inventory. This is a deviation from NEI 99-01 Revision 5. |
| HU2 | 1 | HU2.1 | The third paragraph of the NEI basis has been edited to clarify the significance of the 15-minute duration. If the alarm cannot be verified by redundant Control Room or nearby Fire Panel indications, notification from the field that a fire exists starts the concurrent 15-minute classification and fire suppression clocks. This change is consistent with the manner in which the Control Room and Fire Brigade leaders verify fires. This change is necessary to avoid declaring Unusual Event emergencies for spurious alarms that, due to the sensor location, cannot be verified within 15 minutes of receipt of the alarm.<br>This deviation is based on the EAL FAQ #22 dated September 2010 (ML102030330). This change is necessary to avoid declaring Unusual Event emergencies for spurious alarms that, due to the sensor location, cannot be verified within 15 minutes of receipt of the alarm. This change would eliminate classification due to one indication, a malfunctioning sensor, by requiring additional indications of the existence of a fire. Off-site response organizations have provided feedback during the EAL upgrade process to eliminate |

| NEI | | Ginna EAL/FPB Threshold | Description |
|---|---|---|---|
| IC | Example EAL | | |
| | | | unnecessary classifications that can be easily addressed by plant staff. This change will align the EAL with the rest of the EALs with entry criteria that is based on a number of indications of an event.<br><br>This is a deviation from NEI 99-01 Revision 5. |

# Category R

## Abnormal Rad Release / Radiological Effluent

| NEI IC# | NEI IC Wording and Mode Applicability | Ginna IC#(s) | Ginna IC Wording and Mode Applicability | Difference/Deviation Justification |
|---|---|---|---|---|
| AU1 | Any release of gaseous or liquid radioactivity to the environment greater than 2 times the Radiological Effluent Technical Specifications/ODCM for 60 minutes or longer.<br>MODE: All | RU1 | **ANY** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer<br>MODE: All | The Ginna ODCM limits provide the site-specific Radiological Effluent Technical Specifications. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 60 minutes or longer:<br>(site specific monitor list and threshold values)<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | RU1.1 | **ANY** gaseous or liquid monitor reading > Table R-1 column "UE" for ≥ 60 min. (Note 2)<br>Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | Example EALs 1 and 2 have been combined into a single EAL referencing Table R-1.<br>The NEI phrase "VALID reading on **ANY**…" has been changed to "**ANY**…reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4.<br>The NEI phrase "…of the following radiation monitors greater than the reading shown …(site specific monitor list and threshold values)" has been replaced with "…Gaseous or liquid monitors > Table R-1 column "UE"…" UE, Alert, SAE and GE thresholds for all Ginna continuously monitored gaseous and liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL user. The values shown in Table R-1 column "UE," consistent with the NEI bases, represent two times the ODCM release limits for both liquid and gaseous release.<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 2 | VALID reading on any effluent monitor reading greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | RU1.1 | See RU1.1 above. | Example EALs 1 and 2 have been combined into a single EAL referencing Table R-1. |
| 3 | Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 2 times (site specific RETS values) for 60 minutes or longer.<br>**Note**: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that | RU1.2 | Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x P-9 limits for ≥ 60 min. (Note 2)<br>Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | The Ginna ODCM is the site-specific Radiological Effluent Technical Specifications. The Ginna ODCM limits are specified in Technical Procedure P-9.<br>The NEI phrase "2 times" has been replaced with phrase "2 x" to reduce EAL-user reading burden. The phrases have the same meaning.<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| | the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | | | |
| 4 | VALID reading on perimeter radiation monitoring system reading greater than 0.10 mR/hr above normal* background for 60 minutes or longer. [for sites having telemetered perimeter monitors] | N/A | N/A | Deleted NEI Example EAL #4 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities. |
| 5 | VALID indication on automatic real-time dose assessment capability indicating greater than (site specific value) for 60 minutes or longer. [for sites having such capability] | N/A | N/A | Deleted NEI Example EAL #5 because the plant is not equipped with real-time dose assessment. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities. |

**Table R-1 Effluent Monitor Classification Thresholds**

| Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|
| **Gaseous** | | | | |
| CNMT Vent Noble Gas (R-12) | N/A | N/A | N/A | 7.4E+6 cpm w/ 1 fan<br>5.1E+6 cpm w/ 2 fans |
| CNMT Vent Noble Gas Hi Range (R-12A - 7/9) | 1.8E+2 µC/cc | 1.8E+1 µC/cc | 1.8E+0 µC/cc | N/A |
| Plant Vent Noble Gas (R-14) | N/A | N/A | N/A | 6.0E+5 cpm |
| Plant Vent Noble Gas Hi Range (R-14A - 7/9) | 2.1E+1 µC/cc | 2.1E+0 µC/cc | 2.1E-1 µC/cc | N/A |
| Air Ejector Noble Gas (R-15) | N/A | N/A | N/A | 6.3E+5 cpm |
| Air Ejector Noble Gas Hi Range (R-48) | 5.7E+2 µC/cc | 5.7E+1 µC/cc | 5.7E+0 µC/cc | N/A |
| Main Steam Line (R-31/R-32) | | | | |
| 1 ARV | 5.0E+3 mR/hr | 5.0E+2 mR/hr | 5.0E+1 mR/hr | 8.0E+0 mR/hr |
| 1 Safety | 2.3E+3 mR/hr | 2.3E+2 mR/hr | 2.3E+1 mR/hr | 3.7E+0 mR/hr |
| 2 Safety | 1.1E+3 mR/hr | 1.1E+2 mR/hr | 1.1E+1 mR/hr | N/A |
| 3 Safety | 7.7E+2 mR/hr | 7.7E+1 mR/hr | 7.7E+0 mR/hr | N/A |
| 4 Safety | 5.7E+2 mR/hr | 5.7E+1 mR/hr | 5.7E+0 mR/hr | N/A |
| **Liquid** | | | | |
| Liquid Radwaste Effluent (R-18) | N/A | N/A | N/A | 3.6E+5 cpm with no isolation |
| SFP HX Effluent | | | | |
| (R-20A) | N/A | N/A | N/A | 4.0E+4 cpm |
| (R-20B) | N/A | N/A | N/A | 5.2E+3 cpm |
| Turbine Bldg Flr Drains (R-21) | N/A | N/A | N/A | 5.0E+4 cpm with no isolation |
| Hi Cond Waste (R-22) | N/A | N/A | N/A | 9.2E+4 cpm with no isolation |

| NEI IC# | NEI IC Wording and Mode Applicability | Ginna IC#(s) | Ginna IC Wording and Mode Applicability | Difference/Deviation Justification |
|---|---|---|---|---|
| AU2 | UNPLANNED rise in plant radiation levels<br>MODE: All | RU2 | Unplanned rise in plant radiation levels<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. UNPLANNED water level drop in a reactor refueling pathway as indicated by (site specific level or indication).<br>**AND**<br>b. VALID Area Radiation Monitor reading rise on (site specific list). | RU2.1 | Unplanned water level drop in a reactor refueling pathway as indicated by inability to restore and maintain level > SFP low water level alarm setpoint (Note 3)<br>**AND**<br>Area radiation monitor reading rise on **EITHER**:<br>R-2 Containment<br>**OR**<br>R-5 Spent Fuel Pool<br>Note 3: If loss of water level in the refueling pathway occurs while in Mode 5, 6 or D, consider classification under EALs CU3.1, CU3.2 or CU3.3 | The site specific level or indication of an unplanned water level drop that may result in increased area radiation is the inability to restore and maintain SFP water level > SFP low level alarm setpoint.<br>The NEI term "VALID" has been deleted. All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4.<br>The site specific list of area radiation monitors are monitor R-2 and R-5.<br>Note 3 has been added to the plant EAL wording to ensure subcategory C.3 EALs are reviewed when loss of water shielding above spent fuel adversely affects area radiation levels. |
| 2 | UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels.<br>*Normal can be considered as the highest reading in the past twenty-four hours excluding the | RU2.2 | Unplanned area radiation reading increases by a factor of 1,000 over normal levels | The NEI term "monitor" has been deleted to clarify that radiation readings obtained by portable survey instruments are an acceptable source for assessing this EAL.<br>The NEI term "VALID" has been deleted. All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4.<br>Deleted the asterisk phrase and added the defined phrase to |

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| | current peak value. | | | the EAL Technical Bases: “Normal Levels – As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.” This change implements EAL FAQ #5. |

| NEI IC# | NEI IC Wording | GINNA IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AA1 | Any release of gaseous or liquid radioactivity to the environment greater than 200 times the Radiological Effluent Technical Specifications/ODCM for 15 minutes or longer.<br>MODE: All | RA1 | **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer<br>MODE: All | The Ginna ODCM limits provide the site-specific Radiological Effluent Technical Specifications. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer:<br>(site specific monitor list and threshold values)<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | RA1.1 | **ANY** gaseous monitor reading > Table R-1 column "Alert" for ≥ 15 min. (Note 2)<br>Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | The NEI phrase "VALID reading on **ANY**..." has been changed to "**ANY**...reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4.<br>The NEI phrase "...of the following radiation monitors greater than the reading shown ..." has been replaced with "...gaseous monitors > Table R-1 column "Alert"..."<br>The Ginna radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all Ginna continuously monitored gaseous release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.<br>The value of 1% (10 mrem) of the EPA PAG threshold (in lieu of 200 times the ODCM release rate limit) for gaseous releases is specified to provide a realistic escalation path between the Unusual Event and Site Area Emergency.<br>There are no values specified in Table R-1 for liquid effluent monitors at the Alert or higher classification level. Values corresponding to 200 times ODCM release limits were calculated using the methodology of the ODCM. However, |

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| | | | | since there are no high range monitors associated with the liquid effluent monitoring systems and the 200 times ODCM release limit values are well beyond the upper scale of the instruments, the Alert threshold for liquid releases can only be determined by sample analysis (RA1.2).<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix. |
| 2 | VALID reading on any effluent monitor reading greater than 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | N/A | N/A | Liquid releases at Ginna are the only planned batch releases subject to the discharge permit process. However, there are no values specified in Table R-1 for liquid effluent monitors at the Alert or higher classification level. Values corresponding to 200 times ODCM release limits were calculated using the methodology of the ODCM. However, since there are no high range monitors associated with the liquid effluent monitoring systems and the 200 times ODCM release limit values are well beyond the upper scale of the instruments, the Alert threshold for liquid releases can only be determined by sample analysis (RA1.2). |
| 3 | Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 200 times (site specific RETS values) for 15 minutes or longer.<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but | RA1.2 | Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x P-9 limits for ≥ 15 min.<br>Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the | The Ginna ODCM is the site-specific Radiological Effluent Technical Specifications. The Ginna ODCM limits are specified in Technical Procedure P-9.<br>The NEI phrase "200 times" has been replaced with phrase "200 x" to reduce EAL-user reading burden. The phrases have the same meaning.<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the |

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| | should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | | applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | EAL matrix. |
| 4 | VALID reading on perimeter radiation monitoring system reading greater than 10.0 mR/hr above *normal** background for 15 minutes or longer. [for sites having telemetered perimeter monitors] | N/A | N/A | Deleted NEI Example EAL #4 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities. |
| 5 | VALID indication on automatic real-time dose assessment capability indicating greater than (site specific value) for 15 minutes or longer. [for sites having such capability] | N/A | N/A | Deleted NEI Example EALs #5 because the plant is not equipped with and real-time dose assessment. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AA2 | Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel.<br>MODE: All | RA2 | Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel.<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
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| 1 | A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered. | RA2.2 | A water level drop in a reactor refueling pathway that will result in irradiated fuel becoming uncovered | The terms "reactor refueling cavity, spent fuel pool or fuel transfer canal" have been replaced with the term "reactor refueling pathway" to encompass all three volumes where irradiated fuel may be located. This change implements EAL FAQ #6. |
| 2 | A VALID alarm or (site specific elevated reading) on **ANY** of the following due to damage to irradiated fuel or loss of water level.<br>(site specific radiation monitors) | RA2.1 | Alarm on **ANY** of the following radiation monitors due to damage to irradiated fuel or loss of water level:<br>• R-12 Containment Vent Noble Gas<br>• R-14 Plant Vent Noble Gas<br>• R-2 Containment<br>• R-5 Spent Fuel Pool | The NEI term "VALID" has been deleted. All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4.<br>The EAL provides a site-specific list of radiation monitors applicable to this threshold. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AA3 | Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions<br>MODE: All | RA3 | Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | VALID (site-specific) radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:<br>(Site-specific) list | RA3.1 | Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:<br>Control Room (R-1)<br>**OR**<br>CAS | The NEI term "VALID" has been deleted. All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4.<br>The words "VALID (site-specific) radiation monitor readings GREATER THAN" was replaced with "Dose rates >..." It doesn't matter if the 15 mRem/hr was measured with an ARM or survey instrument; therefore, the term radiation monitor was deleted to not confuse those who may think that only implies a fixed ARM. The symbol ">" means "greater than."<br>The Control Room and CAS are the Ginna areas requiring continuous occupancy to maintain plant safety functions. Since both areas require continuous occupancy, elevated dose rates in any one area could preclude occupancy and, therefore, satisfy the intent of the IC. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AS1 | Off-site dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 100 mrem TEDE or 500 mrem Thyroid CDE for the actual or projected duration of the release.<br>MODE: All MODE: All | RS1 | Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology<br>MODE: All | The NEI abbreviation "mrem" has been replaced with the plant abbreviation "mRem" to agree with units of measure given in the EPA PAGs. This change implements EAL FAQ #8.<br>The phrase "using actual meteorology" has been added to the Ginna IC for consistency with RG1.1 IC wording. This change implements EAL FAQ #9. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | VALID reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:<br>(site specific monitor list and threshold values)<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results. | RS1.1 | **ANY** gaseous monitor reading > Table R-1 column "SAE" for ≥ 15 min. (Note 1)<br>• Do **not** delay declaration awaiting dose assessment results<br>• If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RS1.2)<br>Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time | The NEI phrase "VALID reading on **ANY**..." has been changed to "**ANY** gaseous...reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The word "gaseous" was included consistent with RA1.1, only gaseous monitors listed in Table R-1 are applicable to AS1.<br>The NEI phrase "...of the following... the reading shown..." has been replaced with "...Table R-1 column "SAE"..." The site-specific list is provided in Table R-1.<br>Reference to the NEI note is included in the EAL wording "(Note 1)." Numbering the note facilitates referencing in the EAL matrix.<br>The second and third sentences of the NEI note have been incorporated in the wording of the Ginna EAL for clarification. EAL validation exercises demonstrated the need to emphasize this information in a form other than a note. |

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| 2 | Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary. | RS1.2 | Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary | The NEI abbreviation "mrem" has been replaced with the plant abbreviation "mRem" to agree with units of measure given in the EPA PAGs. This change implements EAL FAQ #8. |
| 3 | VALID perimeter radiation monitoring system reading greater than 100 mR/hr for 15 minutes or longer. [for sites having telemetered perimeter monitors] | N/A | N/A | Deleted NEI Example EAL #3 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities. |
| 4 | Field survey results indicate closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation, at or beyond the site boundary. | RS1.3 | Field survey results indicate closed window dose rates > 100 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary<br>**OR**<br>Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the site boundary | Split the example into two logical conditions separated by the "OR" logical connector for usability.<br>The NEI abbreviation "R" has been replaced with the plant abbreviation "Rem" to agree with units of measure given in the EPA PAGs. This change implements EAL FAQ #8.<br>The NEI phrase "one hour" has been abbreviated "1 hr" to reduce EAL-user reading burden. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AG1 | Off-site dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 1000 mrem TEDE or 5000 mrem Thyroid CDE for the actual or projected duration of the release using actual meteorology.<br>MODE: All | RG1 | Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology<br>MODE: All | None. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer:<br>(site specific monitor list and threshold values)<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results. | RG1.1 | **ANY** gaseous monitor reading > Table R-1 column "GE" for ≥ 15 min. (Note 1)<br>• Do **not** delay declaration awaiting dose assessment results<br>• If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RG1.2)<br>Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time | The NEI phrase "VALID reading on **ANY**..." has been changed to "**ANY** gaseous...reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The word "gaseous" was included consistent with RA1.1, only gaseous monitors listed in Table R-1 are applicable to AG1.<br>The NEI phrase "...of the following... the reading shown..." has been replaced with "...Table R-1 column "GE"..." The site-specific list is provided in Table R-1.<br>Reference to the NEI note is included in the EAL wording "(Note 1)." Numbering the note facilitates referencing in the EAL matrix.<br>The second and third sentences of the NEI note have been incorporated in the wording of the Ginna EAL for clarification. EAL validation exercises demonstrated the need to emphasize this information in a form other than a note. |

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| 2 | Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary. | RG1.2 | Dose assessment using actual meteorology indicates doses > 1,000 mRem TEDE or 5,000 mRem thyroid CDE at or beyond the site boundary | None |
| 3 | VALID perimeter radiation monitoring system reading greater than 1000 mR/hr for 15 minutes or longer. [for sites having telemetered perimeter monitors] | N/A | N/A | Deleted NEI Example EAL #3 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities. |
| 4 | Field survey results indicate closed window dose rates greater than 1000 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for one hour of inhalation, at or beyond site boundary. | RG1.3 | Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary<br>**OR**<br>Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the site boundary | Split the example into two logical conditions separated by the "OR" logical connector for usability.<br>The NEI abbreviation "R" has been replaced with the plant abbreviation "Rem" to agree with units of measure given in the EPA PAGs. This change implements EAL FAQ #8.<br>The NEI phrase "one hour" has been abbreviated "1 hr" to reduce EAL-user reading burden. |

# Category C

# Cold Shutdown / Refueling System Malfunction

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU1 | RCS Leakage<br>MODE: Cold Shutdown | CU3 | RCS Leakage<br>MODE: 5 - Cold Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>1. RCS leakage results in the inability to maintain or restore RPV level greater than (site specific low level RPS actuation setpoint) for 15 minutes or longer. [*BWR*]<br>1. RCS leakage results in the inability to maintain or restore level within (site specific pressurizer or RCS/RPV level target band) for 15 minutes or longer. [*PWR*] | CU3.1 | RCS leakage results in the inability to maintain or restore RCS level within the target band established by procedure for ≥ 15 min. (Note 4)<br>Note 4: The ED 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 | The BWR portion of the NEI EAL has not been implemented because Ginna is a PWR.<br>There is no specific Pressurizer level relevant to RCS level control in the Cold Shutdown mode, only as specified by procedure for the given configuration.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU2 | UNPLANNED loss of RCS/RPV inventory.<br>MODE: Refueling | CU3 | RCS leakage<br>MODE: 6 - Refuel | IC wording aligned with NEI IC CU1 to support grouping NEI IC CU1 and CU2 EALs under the same subcategory. There is no fundamental difference between an unplanned loss of RCS inventory and RCS leakage. This change implements EAL FAQ #41. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
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| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>Unplanned RCS/RPV level drop as indicated by either of the following:<br>• RCS/RPV water level drop below the RPV flange for 15 minutes or longer when the RCS/RPV level band is established above the RPV flange.<br>• RCS/RPV water level drop below the RCS level band for 15 minutes or longer when the RCS/RPV level band is established below the RPV flange. | CU3.2 | Unplanned RCS level drop below **EITHER** of the following for ≥ 15 min. (Note 4):<br>Reactor Vessel flange (84 in. on loop level indicators) (when the level band is established above the flange)<br>**OR**<br>RCS level target band established by procedure (when the level band is established below the flange)<br>Note 4: The ED 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. | The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.<br>The NEI phrase "RPV flange" has been replaced with "Reactor Vessel flange" to use terminology commonly accepted at PWRs.<br>In the second bullet, the NEI phrase "RCS level band" has been replaced with "RCS level target band" for consistency with terminology used in EAL CU3.1.<br>The NEI introductory clause and the two NEI bulleted conditions have been reworded for clarification. |
| 2 | RCS/RPV level cannot be | CU3.3 | RCS level **cannot** be monitored | The NEI abbreviation "RCS/RPV" has been changed to "RCS" to |

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| | monitored with a loss of RCS/RPV inventory as indicated by an unexplained level rise in (site specific sump or tank). | | with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage | use terminology commonly accepted at PWRs.<br>The NEI phrase "(site-specific sump or tank)" has been replaced with "**ANY** Table C-2 sump / tank attributable to RCS leakage" for clarification. The list of sumps and tanks is too large to include within the wording of the EAL and maintain readability. Table C-2 contains the site-specific list of sumps and tanks. |

| **Table C-2 RCS Leakage Indications** |
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| • Containment Sump A<br>• Containment Sump B<br>• Auxiliary Building Sump Tank<br>• Reactor Coolant Drain Tank (RCDT) |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU3 | AC power capability to emergency busses reduced to a single power source for 15 minutes or longer such that any additional single failure would result in station blackout.<br>MODE: Cold Shutdown, Refueling | CU1 | AC power capability to 480V safeguards buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 480V safeguards bus power<br>MODE: 5 - Cold Shutdown, 6 - Refuel, D - Defueled | "...emergency busses..." replaced with "...480V safeguards buses..." as the site specific terminology for emergency buses.<br>"...station blackout." replaced with "...a complete loss of all 480V safeguards bus power" as this describes the intended condition for Ginna.<br>Added "D – Defueled" to the mode applicability to correct omission in NEI 99-01. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>a. AC power capability to (site specific emergency busses) reduced to a single power source for 15 minutes or longer.<br>**AND**<br>b. Any additional single power source failure will result in station blackout. | CU1.1 | AC power capability to 480V safeguards buses reduced to a single power source, Table C-1, for ≥ 15 min. (Note 4)<br>**AND**<br>**Any** additional single power source failure will result in a complete loss of all 480V safeguards bus power<br>Note 4: The ED 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. | 480V safeguards buses are the Ginna emergency buses.<br>Table C-1 provides a list of Ginna onsite and offsite AC power sources in cold conditions.<br>The NEI phrase "..station blackout" has been replaced with "... complete loss of all 480V safeguards bus power" as this describes the intended condition for Ginna. This change implements EAL FAQ #36.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| Table C-1 AC Power Sources | |
|---|---|
| Onsite | • EDG 1A (Safeguard train A, Buses 14 & 18)<br>• EDG 1B (Safeguard train B, Buses 16 & 17) |
| Offsite | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU4 | UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV.<br>MODE: Cold Shutdown, Refueling | CU4 | Unplanned loss of decay heat removal capability<br>MODE: 5 - Cold Shutdown, 6 - Refuel | The NEI phrase "with irradiated fuel in the RPV" has been deleted to implement EAL FAQ #11. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit. | CU4.1 | Unplanned event results in RCS temperature > 200°F | The NEI phrase "...exceeding the Technical Specification cold shutdown temperature limit" has been replaced with "> 200°F." >200°F is the Technical Specification cold shutdown temperature limit and is specified in the EAL instead of the NEI wording to reduce EAL-user reading burden. |
| 2 | Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>Loss of all RCS temperature and RCS/RPV level indication for 15 minutes or longer. | CU4.2 | Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 4)<br>Note 4: The ED 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. | The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU6 | Loss of all On-site or Off-site communications capabilities.<br>MODE: Cold Shutdown, Refueling, Defueled | CU5 | Loss of **all** onsite or offsite communications capabilities<br>MODE: 5 - Cold Shutdown, 6 - Refuel, D - Defueled | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss of all of the following on-site communication methods affecting the ability to perform routine operations:<br>(site specific list of communications methods) | CU5.1 | Loss of **all** Table C-5 onsite (internal) communication methods affecting the ability to perform routine operations<br>**OR**<br>Loss of **all** Table C-5 offsite (external) communication methods affecting the ability to perform offsite notifications | CU5.1 implements Example EALs #1 and #2. These were combined for improved usability.<br>The NEI example EALs specify site-specific lists of onsite and offsite communications methods. The Ginna EAL lists these methods in Table C-5 because the number of communications methods is too long to include within the text of the EAL.<br>The adjectives "(internal)" and "(external)" have been added to the Ginna EAL for clarification. The terms "onsite/offsite" could be interpreted as the location in which the communication originates instead of the location to which communication is directed. |
| 2 | Loss of all of the following off-site communication methods affecting the ability to perform offsite notifications:<br>(site specific list of communications methods) | | | |

| Table C-5 Communications Systems | | |
|---|---|---|
| **System** | **Onsite** (internal) | **Offsite** (external) |
| Commercial phone system | X | X |
| Direct Dial "POTS" Lines (Blue Phones) | X | X |
| Plant Page Party system | X | |
| Radios/Walkie Talkies | X | |
| FTS 2001 telephone system (ENS, HPN) | | X |
| Control Room Hard Wired Satellite Phone | | X |
| Control Room Emergency Cell Phone | | X |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU7 | Loss of required DC power for 15 minutes or longer.<br>MODE: Cold Shutdown, Refueling | CU2 | Loss of **required** DC power for ≥ 15 min.<br>MODE: 5 - Cold Shutdown, 6 - Refuel | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br><br>Less than (site specific bus voltage indication) on required (site specific Vital DC busses) for 15 minutes or longer. | CU2.1 | < 108 VDC on **required** 125 VDC buses for ≥ 15 min. (Note 4)<br>Note 4: The ED 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. | "108 VDC" is the site-specific bus voltage indication.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU8 | Inadvertent Criticality<br>MODE: Cold Shutdown, Refueling | CU6 | Inadvertent criticality<br>MODE: 5 - Cold Shutdown, 6 - Refuel | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | UNPLANNED sustained positive period observed on nuclear instrumentation. (BWR) | N/A | N/A | NEI Example EAL #1 has not been implemented because it applies only to BWR plants. Ginna is a PWR. PWRs are not equipped with period meters. |
| 2 | UNPLANNED sustained positive startup rate observed on nuclear instrumentation. (PWR) | CU6.1 | An unplanned sustained positive startup rate observed on nuclear instrumentation | None |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CA1 | Loss of RCS/RPV inventory.<br>MODE: Cold Shutdown, Refueling | CA3 | Loss of RCS inventory<br>MODE: 5 - Cold Shutdown, 6 - Refuel | The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss of RCS/RPV inventory as indicated by level less than (site specific level).<br>[Low-Low ECCS actuation setpoint / Level 2 (BWR)]<br>[Bottom ID of the RCS loop (PWR)] | CA3.1 | Loss of inventory as indicated by RCS water level < 0 in.<br>**OR**<br>RCS level **cannot** be monitored for ≥ 15 min. with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage (Note 4)<br>Note 4: The ED 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. | CA3.1 implements Example EALs #1 and #2. These were combined for improved usability.<br>The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.<br>The bottom ID of the RCS hot leg is indicated by RCS loop level indicators at 0 in.<br>Ginna is a PWR and is not equipped with the BWR low-low ECCS actuation setpoint.<br>The NEI phrase "(site-specific sump or tank)" has been replaced with "**ANY** Table C-2 sump / tank attributable to RCS leakage" for clarification. The list of sumps and tanks is too large to include within the wording of the EAL and maintain readability. Table C-2 contains the site-specific list of sumps and tanks. Consistent with the developers guidance that the source of the leakage needs to be evaluated against other sources, clarification was added with the words "attributable to RCS leakage".<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |
| 2 | Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>RCS/RPV level cannot be monitored for 15 minutes or longer with a loss of RCS/RPV inventory as indicated by an unexplained level rise in (site specific sump or tank). | | | |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CA3 | Loss of all Off-site and all On-Site AC power to emergency busses for 15 minutes or longer.<br>MODE: Cold Shutdown, Refueling, Defueled | CA1 | Loss of **all** offsite and **all** onsite AC power to 480V safeguards buses for ≥ 15 min.<br>MODE: 5 - Cold Shutdown, 6 - Refuel, D - Defueled | "...emergency busses..." replaced with "...480V safeguards buses..." as the site specific terminology for emergency buses. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>Loss of all Off-Site and all On-Site AC Power to (site specific emergency busses) for 15 minutes or longer. | CA1.1 | Loss of **all** offsite and **all** onsite AC power, Table C-1, to 480V safeguards buses for ≥ 15 min. (Note 4)<br>Note 4: The ED 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. | 480V safeguards buses are the Ginna emergency buses.<br>Table C-1 provides a list of Ginna onsite and offsite AC power sources in cold conditions.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CA4 | Inability to maintain plant in cold shutdown.<br>MODE: Cold Shutdown, Refueling | CA4 | Inability to maintain plant in cold shutdown<br>MODE: 5 - Cold Shutdown, 6 - Refuel | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | An UNPLANNED event results in RCS temperature greater than (site specific Technical Specification cold shutdown temperature limit) for greater than the specified duration on table. | CA4.1 | An unplanned event results in **EITHER**:<br>RCS temperature > 200°F for > Table C-4 duration<br>**OR**<br>RCS pressure increase > 10 psi due to an unplanned loss of decay heat removal capability (this condition is **not** applicable in solid plant conditions) | CA4.1 implements NEI EALs #1 and #2. The NEI example EALs have been combined for simplification.<br>The NEI phrase "...exceeding the Technical Specification cold shutdown temperature limit" has been replaced with "> 200°F." 200°F is the Technical Specification cold shutdown temperature limit.<br>NEI criteria associated with RCS temperature exceeding the Technical Specification cold shutdown temperature limit are given in Table C-4.<br>The NEI phrase "An UNPLANNED event results in RCS pressure increase greater than 10 psi due to a loss of RCS cooling" has been changed to "RCS pressure increase > 10 psi due to an unplanned loss of decay heat removal capability" for clarification. This change implements EAL FAQ #13.<br>Pressure indicator PI-420 Rx Clnt Loop Lo Rng Press is capable of measuring pressure changes of 10 psig. |
| 2 | An UNPLANNED event results in RCS pressure increase greater than 10 psi due to a loss of RCS cooling. (PWR-This EAL does not apply in Solid Plant conditions.) | | | |

| Table: RCS Reheat Duration Thresholds | | |
|---|---|---|
| **RCS** | **Containment Closure** | **Duration** |
| Intact (but not RCS Reduced Inventory [PWR]) | N/A | 60 minutes* |
| Not intact or RCS Reduced Inventory *(PWR)* | Established | 20 minutes* |
| | Not Established | 0 minutes |
| * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable. | | |

| Table C-4 RCS Reheat Duration Thresholds | | |
|---|---|---|
| RCS Status | Containment Closure Status | Duration |
| Intact **AND not** reduced inventory | N/A | 60 min.* |
| **Not** intact **OR** reduced inventory | Established | 20 min.* |
| | **Not** established | 0 min. |

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

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CS1 | Loss of RCS/RPV inventory affecting core decay heat removal capability<br>MODE: Cold Shutdown, Refueling | CS3 | Loss of RCS inventory affecting core decay heat removal capability<br>MODE: 5 - Cold Shutdown, 6 - Refuel | The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | With CONTAINMENT CLOSURE not established, RCS/RPV level less than (site specific level).<br>[6" below the bottom ID of the RCS loop (PWR)]<br>[6" below the low-low ECCS actuation setpoint (BWR)] | N/A | N/A | Ginna cannot measure reactor vessel level below the bottom of the RCS hot leg. Consistent with the NEI 99-01 CS1 developers guidance this example EAL is not implemented. |
| 2 | With CONTAINMENT CLOSURE established, RCS/RPV level less than (site specific level for TOAF). | N/A | N/A | Ginna cannot measure reactor vessel level below the bottom of the RCS hot leg. Consistent with the NEI 99-01 CS1 developers guidance this example EAL is not implemented. |
| 3 | Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>RCS/RPV level cannot be monitored for 30 minutes or longer with a loss of RCS/RPV | CS3.1 | RCS level **cannot** be monitored with a loss of RCS inventory as indicated by **ANY** of the following for ≥ 30 min. (Note 4):<br>• Containment radiation R-29 or R-30 > 1.0E+02 R/hr<br>• Erratic Source Range Nuclear Instrumentation indication | The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.<br>"for ≥ 30 min." has been placed at the end of the sentence for consistency with NEI CG1 Example EAL #2 and Ginna EAL CG3.1.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.<br>Containment radiation is indicated on R-29 or R-30. The containment radiation monitors high alarm is set at 1.0E+02 R/hr. The 1.0E+02 R/hr setpoint has been selected to be operationally |

| | | | | |
|---|---|---|---|---|
| | inventory as indicated by ANY of the following:<br>• (Site specific radiation monitor) reading greater than (site specific value).<br>• Erratic Source Range Monitor Indication.<br>• Unexplained level rise in (site specific sump or tank). | | • Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage<br>Note 4: The ED 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. | significant and above that expected under normal plant conditions while in the Refuel mode.<br>The NEI phrase "(site-specific sump or tank)" has been replaced with "**ANY** Table C-2 sump / tank attributable to RCS leakage" for clarification. The list of sumps and tanks is too large to include within the wording of the EAL and maintain readability. Table C-2 contains the site-specific list of sumps and tanks. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CG1 | Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged.<br>MODE: Cold Shutdown, Refueling | CG3 | Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged<br>MODE: 5 - Cold Shutdown, 6 - Refuel | The NEI abbreviation "RCS/RPV" has been changed to "Reactor Vessel" to use terminology commonly accepted at PWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>a. RCS/RPV level less than (site specific level for TOAF) for 30 minutes or longer.<br>**AND**<br>b. **ANY** containment challenge indication (see Table): | N/A | N/A | Ginna cannot measure reactor vessel level below the bottom of the RCS hot leg. Consistent with the NEI 99-01 developers guidance this example EAL is not implemented. |
| 2 | a. RCS/RPV level cannot be monitored with core uncovery indicated by **ANY** of the following for 30 minutes or longer.<br>• (Site specific radiation monitor) reading greater than (site specific setpoint).<br>• Erratic source range | CG3.1 | RCS level **cannot** be monitored with core uncovery indicated by **ANY** of the following for ≥ 30 min. (Note 4):<br>• Containment radiation R-29 or R-30 > 1.0E+02 R/hr<br>• Erratic Source Range Nuclear Instrumentation | The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.<br>Containment radiation is indicated on R-29 or R-30. The containment radiation monitors high alarm is set at 1.0E+02 R/hr. The 1.0E+02 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions |

| | | | | |
|---|---|---|---|---|
| | monitor indication<br>• UNPLANNED level rise in (site specific sump or tank).<br>• *[Other site specific indications]*<br>**AND**<br>b. **ANY** containment challenge indication (see Table): | | indication<br>• Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage<br>**AND**<br>**Any** Containment Challenge Indication, Table C-3<br>Note 4: The ED 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 | while in the Refuel mode.<br>The NEI term "UNPLANNED" has been changed to "Unexplained" for consistency with NEI IC CS1 Example EAL #3f.<br>The NEI phrase "(site-specific sump or tank)" has been replaced with "**ANY** Table C-2 sump / tank attributable to RCS leakage" for clarification. The list of sumps and tanks is too large to include within the wording of the EAL and maintain readability. Table C-2 contains the site-specific list of sumps and tanks.<br>Other site-specific indications of core uncovery could not be identified.<br>Table C-3 lists the Containment Challenge indications. "Secondary containment radiation monitor reading above" has not been incorporated in Table C-3 because Ginna is a PWR and not equipped with a secondary containment. |

| **Table C-2 RCS Leakage Indications** |
|---|
| • Containment Sump A |
| • Containment Sump B |
| • Auxiliary Building Sump Tank |
| • Reactor Coolant Drain Tank (RCDT) |

| Table C-3 Containment Challenge Indications |
|---|
| • Containment closure **not** established<br>• Hydrogen concentration in Containment ≥ 4%<br>• Unplanned rise in Containment pressure |

# Category D

# Permanently Defueled Station Malfunction

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| D-AU1<br>D-AU2<br>D-SU1<br>D-HU1<br>D-HU2<br>D-HU3<br>D-AA1<br>D-AA2<br>D-HA1<br>D-HA2 | Recognition Category D<br>Permanently Defueled Station Malfunction | N/A | N/A | NEI Recognition Category D ICs and EALs are applicable only to permanently defueled stations. Ginna is not a defueled station. |

# Category E

## Events Related to Independent Spent Fuel Storage Installations

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| E-HU1 | Damage to a loaded cask CONFINEMENT BOUNDARY<br>MODE: Not applicable | EU1 | Damage to a loaded cask confinement boundary<br>MODE: Not applicable | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Damage to a loaded cask confinement BOUNDARY | EU1.1 | Damage to a loaded cask confinement boundary | None |

# Category F

# Fission Product Barrier Degradation

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FU1 | ANY Loss or ANY Potential Loss of Containment<br>MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FU1 | **ANY** loss or **ANY** potential loss of Containment<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | ANY Loss or ANY Potential Loss of Containment | FU1.1 | **ANY** loss or **ANY** potential loss of Containment (Table F-1) | Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-3. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FA1 | ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS<br>MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FA1 | **ANY** loss or **ANY** potential loss of either Fuel Clad or RCS<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS | FA1.1 | **ANY** loss or **ANY** potential loss of either Fuel Clad or RCS (Table F-1) | Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-3. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FS1 | Loss or Potential Loss of ANY Two Barriers<br>MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FS1 | Loss or potential loss of **ANY** two barriers<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss or Potential Loss of ANY Two Barriers | FS1.1 | Loss or potential loss of **ANY** two barriers (Table F-1) | Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-3. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FG1 | Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier<br>MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FG1 | Loss of **ANY** two barriers and loss or potential loss of the third barrier<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier | FG1.1 | Loss of **ANY** two barriers<br>**AND**<br>Loss or potential loss of the third barrier (Table F-1) | Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-3. |

| NEI Ex. EAL # | NEI Table 5-F-1 Notes | Ginna EAL # | Ginna Bases Discussion | Difference/Deviation Justification |
|---|---|---|---|---|
| N/A | **NOTES**<br><br>The logic used for these initiating conditions reflects the following considerations:<br><br>• The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier (See Sections 3.4 and 3.8). NOUE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.<br><br>• At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" EALs existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need | FU1.1<br>FA1.1<br>FS1.1<br>FG1.1 | The logic used for Category F EALs reflects the following considerations:<br><br>• The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. UE EALs associated with RCS and Fuel Clad Barriers are addressed under Category S.<br><br>• At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" thresholds existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" thresholds existed, the ED would have more assurance that there was no immediate need to escalate to a General | First bullet: The NEI parenthetical phrase "See Sections 3.4 and 3.8" has been deleted because it refers to NEI EAL developmental information.<br><br>First bullet: The NEI acronym "NOUE" has been implemented as "UE" for simplification. The NEI abbreviation "ICs" has been changed to "EALs" for clarification.<br><br>Second bullet: The NEI abbreviation "EALs" has been changed to "thresholds" for clarification.<br><br>The second sentence in the fourth bullet of the NEI notes "When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications" has been deleted to implement EAL FAQ #14. |

| | | | | |
|---|---|---|---|---|
| | to escalate to a General Emergency.<br>• The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.<br>• The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications. | | Emergency.<br>• The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.<br>• The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. | |

| Table F-1 Fission Product Barrier Matrix | | | | | | |
|---|---|---|---|---|---|---|
| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Containment Barrier | |
| Category | Loss | Potential Loss | Loss | Potential Loss | Loss | Potential Loss |
| **A** **CSFST** | 1. **RED** path condition exists F-0.2 Core Cooling | 1. **ORANGE** path condition exists F-0.2 Core Cooling<br>2. **RED** path condition exists F-0.3 Heat Sink and heat sink is required | None | 1. **RED** path condition exists F-0.4 Integrity<br>2. **RED** path condition exists F-0.3 Heat Sink and heat sink is required | None | 1. **RED** path condition exists F-0.5 Containment |
| **B** **Core Exit TCs** | 2. Core Exit TCs ≥ 1,200°F | 3. Core Exit TCs ≥ 700°F | None | None | None | 2. Core Exit TCs **cannot** be restored < 1,200°F within 15 min.<br>3. Core Exit TCs ≥ 700°F **AND** RVLIS level **cannot** be restored > 52% [ > 55% adverse CNMT] with no RCPs running within 15 min. |
| **C** **Inventory** | None | 4. RVLIS level ≤ 52% [ ≤ 55% adverse CNMT] **OR** At least one RCP running RVLIS fluid fraction ≤ 66% | 1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< EOP Fig. MIN SUBCOOLING)<br>2. Ruptured S/G results in an ECCS (SI) actuation | 3. RCS leak rate > 50 gpm with letdown isolated | 1. A containment pressure rise followed by a rapid unexplained drop in containment pressure<br>2. Containment pressure or sump level response **not** consistent with LOCA conditions<br>3. Ruptured S/G is also faulted outside of containment<br>4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable prolonged steam release from affected S/G to the environment | 4. Containment pressure ≥ 60 psig and rising<br>5. Containment hydrogen concentration ≥ 4%<br>6. Containment pressure ≥ 28 psig and < two CRFC units and one CS pump operating per design |
| **D** **Radiation / Coolant Activity** | 3. Containment radiation monitor R-29/R-30 reading > 1.0E+02 R/hr<br>4. Valid Letdown Monitor (R-9) reading ≥ 24,000 mRad/hr<br>5. Coolant activity >300 µCi/gm dose equivalent I-131 | None | 3. Containment radiation monitor R-29/R-30 reading > 1.0E+01 R/hr | None | None | 7. Containment radiation monitor R-29/R-30 reading > 1.0E+03 R/hr |
| **E** **Isolation Status** | None | None | None | None | 5. Failure of **all** valves in **ANY one** line to close **AND** Direct downstream pathway to the environment exists after containment isolation signal | None |
| **F** **Judgment** | 6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier | 5. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier | 4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier | 4. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier | 6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the containment barrier | 8. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the containment barrier |

**Fuel Clad Fission Product Barrier Degradation Thresholds**

| NEI FPB# | NEI Threshold Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FC Loss 1 | **Critical Safety Function Status**<br>A. Core-Cooling Red Entry Conditions Met. | FC Loss A.1 | **RED** path condition exists F-0.2 Core Cooling | The NEI phrase "Core-Cooling Red Entry Conditions Met" has been changed to "**RED** path condition exists F-0.2 Core Cooling." Procedure F-0.2 is the core cooling CSFST.<br>The NEI phrase "Entry Conditions Met" has been changed to "condition exists" for consistency with terminology used by Ginna operators when using the EOPs. |
| FC Loss 2 | **Primary Coolant Activity Level**<br>A. Coolant activity greater than (site specific value). | FC Loss D.5 | Coolant activity >300 μCi/gm dose equivalent I-131 | 300 μCi/gm dose equivalent I-131 is the site-specific value for coolant activity.<br>The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. |
| FC Loss 3 | **Core Exit Thermocouple Readings**<br>A. Core exit thermocouples reading greater than (site specific degree F). | FC Loss B.2 | Core Exit TCs ≥ 1,200°F | TCs is the Ginna equivalent of NEI "thermocouple readings."<br>The TC value specifies "≥" vs. ">" consistent with F-0.2 Core Cooling RED path entry conditions.<br>The NEI word "degree" has been replaced with "°" to reduce EAL-user reading burden. The symbol "°" is commonly understood to mean "degree."<br>1,200°F is the Ginna specific temperature corresponding to significant core exit superheating and core uncovery. |
| FC Loss 4 | **Reactor Vessel Water Level**<br>Not Applicable | N/A | N/A | N/A |
| FC Loss 5 | **Not Applicable**<br>Not Applicable | N/A | N/A | N/A |

| NEI FPB# | NEI Threshold Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FC Loss 6 | **Containment Radiation Monitoring**<br>A. Containment radiation monitor reading greater than (site specific value). | FC Loss D.3 | Containment radiation monitor R-29/R-30 reading > 1.0E+02 R/hr | 1.0E+02 R/hr is the site-specific containment rad monitor reading. |
| FC Loss 7 | **Other (Site-Specific) Indications**<br>A. (Site-specific ) as applicable | FC Loss D.4 | Valid Letdown Monitor (R-9) reading ≥ 24,000 mRad/hr | A Letdown Line Monitor reading of 24,000 mRad/hr represents fuel clad damage of approximately 5% corresponding to the reactor coolant activity fuel Clad loss threshold of 300 μCi/gm dose equivalent I-131 |
| FC Loss 8 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier. | FC Loss F.6 | **ANY** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier | None |
| FC P-Loss 1 | **Critical Safety Function Status**<br>A. Core Cooling-Orange Entry Conditions Met.<br>**OR**<br>B. Heat Sink-Red Entry Conditions Met. | FC P-Loss A.1 | **ORANGE** path condition exists F-0.2 Core Cooling | The NEI phrase "Core Cooling-Orange Entry Conditions Met" has been changed to "**ORANGE** path condition exists F-0.2 Core Cooling." Procedure F-0.2 is the core cooling CSFST.<br>The NEI phrase "Entry Conditions Met" has been changed to "condition exists" for consistency with terminology used by Ginna operators when using the EOPs. |
| | | FC P-Loss A.2 | **RED** path condition exists F-0.3 Heat Sink and heat sink is required | The NEI phrase "Heat Sink-Red Entry Conditions Met" has been changed to "**RED** path condition exists F-0.3 Heat Sink." Procedure F-0.3 is the heat sink CSFST.<br>The NEI phrase "Entry Conditions Met" has been changed to "condition exists" for consistency with terminology used by Ginna operators when using the EOPs.<br>The phrase "and heat sink is required" has been added to the Ginna threshold to implement EAL FAQ #18. |

| NEI FPB# | NEI Threshold Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FC P-Loss 2 | **Primary Coolant Activity Level**<br>Not Applicable | N/A | N/A | N/A |
| FC P-Loss 3 | **Core Exit Thermocouple Readings**<br>A.. Core exit thermocouples reading greater than (site specific degree F). | FC P-Loss B.3 | Core Exit TCs ≥ 700°F | TCs is the Ginna equivalent of NEI "thermocouple readings."<br>The TC value specifies "≥" vs. ">" consistent with F-0.2 Core Cooling Orange path entry conditions.<br>The NEI word "degree" has been replaced with "°" to reduce EAL-user reading burden. The symbol "°" is commonly understood to mean "degree."<br>700°F is the Ginna specific temperature corresponding to significant core exit superheating. |
| FC P-Loss 4 | **Reactor Vessel Water Level**<br>A. RCS/RPV level less than (site specific level for TOAF). | FC P-Loss C.4 | RVLIS level ≤ 52% [≤ 55% adverse CNMT]<br>**OR**<br>At least one RCP running RVLIS fluid fraction ≤ 66% | The NEI phrase "RCS/RPV" has been replaced with "RVLIS" to use terminology consistent with the Ginna EOPs.<br>The Reactor Vessel water level threshold is used in the EOPs to signal core uncovery and is, therefore, indication of inadequate coolant inventory. If the RVLIS indication drops to 52% [≤ 55% adverse CNMT] or with at least one RCP running RVLIS fluid fraction ≤ 66%, a core covered condition cannot be confirmed. According to the Core Cooling-ORANGE path, this water level indicates subcooling has been lost and that some fuel clad damage may occur. |
| FC P-Loss 5 | **Not Applicable**<br>Not Applicable | N/A | N/A | N/A |
| FC P-Loss 6 | **Containment Radiation Monitoring**<br>Not Applicable | N/A | N/A | N/A |

| NEI FPB# | NEI Threshold Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FC P-Loss 7 | **Other (Site-Specific) Indications**<br>A. (Site-specific) as applicable | N/A | None | Other site-specific indications of fuel clad potential loss have not been identified. |
| FC P-Loss 8 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier. | FC P-Loss F.5 | **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier | None |

**RCS Fission Product Barrier Degradation Thresholds**

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| RCS Loss 1 | **Critical Safety Function Status**<br>Not Applicable | N/A | N/A | None |
| RCS Loss 2 | **RCS Leak Rate**<br>A. RCS leak rate greater than available makeup capacity as indicated by a loss of RCS subcooling. | RCS Loss C.1 | RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< EOP Fig. MIN SUBCOOLING) | Critical Safety Function Status Trees (CSFST) Core Cooling indicates that if subcooling margin based on core exit TCs is in the Inadequate Subcooling Region of EOP Fig. MIN SUBCOOLING, a loss of RCS subcooling has occurred. The loss of subcooling as a result of inability to establish RCS heat transfer to the ultimate heat sink is indicative of potential losses of the Fuel Cladding and RCS barriers. |
| RCS Loss 3 | **Not Applicable**<br>Not Applicable | N/A | N/A | None |
| RCS Loss 4 | **SG Tube Rupture**<br>A. RUPTURED SG results in an ECCS (SI) actuation. | RCS Loss C.2 | Ruptured S/G results in an ECCS (SI) actuation | None |
| RCS Loss 5 | **Not Applicable**<br>Not Applicable | N/A | N/A | None |
| RCS Loss 6 | **Containment Radiation Monitoring**<br>A. Containment radiation monitor reading greater than (site specific value). | RCS Loss D.3 | Containment radiation monitor R-29/R-30 reading > 1.0E+01 R/hr | R-29 & R-30 alert alarms at 1.0E+01 R/hr, indicative of a significant RCS breach (LOCA) in containment (~0.1% gap activity). |
| RCS Loss 7 | **Other (Site-Specific) Indications**<br>A. (Site-specific) as applicable | N/A | None | Other site-specific indications of RCS loss have not been identified. |

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| RCS Loss 8 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier. | RCS Loss F.4 | **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier | None |
| RCS P-Loss 1 | **Critical Safety Function Status**<br>A. RCS Integrity-Red Entry Conditions Met.<br>**OR**<br>B. Heat Sink-Red Entry Conditions Met. | RCS P-Loss A.1 | **RED** path condition exists F-0.4 Integrity | The NEI phrase "RCS Integrity-Red Entry Conditions Met" has been changed to "**RED** path condition exists F-0.4 Integrity." Procedure F-0.4 is the integrity CSFST.<br>The NEI phrase "Entry Conditions Met" has been changed to "condition exists" for consistency with terminology used by Ginna operators when using the EOPs. |
| | | RCS P-Loss A.2 | **RED** path condition exists F-0.3 Heat Sink and heat sink is required | The NEI phrase "Heat Sink-Red Entry Conditions Met" has been changed to "**RED** path condition exists F-0.3 Heat Sink." Procedure F-0.3 is the heat sink CSFST.<br>The NEI phrase "Entry Conditions Met" has been changed to "condition exists" for consistency with terminology used by Ginna operators when using the EOPs.<br>The phrase "and heat sink is required" has been added to the Ginna threshold to implement EAL FAQ #18. |
| RCS P-Loss 2 | **RCS Leak Rate**<br>A. RCS leak rate indicated greater than (site specific capacity of one charging pump in the normal charging mode) with Letdown isolated. | RCS P-Loss C.3 | RCS leak rate $\geq$ 50 gpm with letdown isolated | The CVCS includes three positive displacement horizontal pumps with a capacity of 46 gpm each. The single charging pump capacity is rounded up to 50 gpm for this threshold consistent with the generic developers guidance for plants with low capacity charging pumps and clearly signals that operation of more than one charging pump is needed. |
| RCS P-Loss 3 | **Not Applicable**<br>Not Applicable | N/A | N/A | N/A |
| RCS P-Loss 4 | **SG Tube Rupture**<br>Not Applicable | N/A | N/A | N/A |

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| RCS P-Loss 5 | **Not Applicable**<br>Not Applicable | N/A | N/A | N/A |
| RCS P-Loss 6 | **Containment Radiation Monitoring**<br>Not Applicable | N/A | N/A | N/A |
| RCS P-Loss 7 | **Other (Site-Specific) Indications**<br>A. (Site-specific) as applicable | N/A | None | Other site-specific indications of RCS potential loss have not been identified. |
| RCS P-Loss 8 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier. | RCS P-Loss F.4 | **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier | None |

**Containment Fission Product Barrier Degradation Thresholds**

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CNMT Loss 1 | **Critical Safety Function Status**<br>Not Applicable | N/A | N/A | None |
| CNMT Loss 2 | **Containment Pressure**<br>A. A containment pressure rise followed by a rapid unexplained drop in containment pressure.<br>**OR**<br>B. Containment pressure or sump level response not consistent with LOCA conditions. | CNMT Loss C.1 | A containment pressure rise followed by a rapid unexplained drop in containment pressure | The NEI threshold has been divided into two Ginna thresholds to improve clarity. |
| | | CNMT Loss C.2 | Containment pressure or sump level response **not** consistent with LOCA conditions | The NEI threshold has been divided into two Ginna thresholds to improve clarity. |
| CNMT Loss 3 | **Core Exit Thermocouple Readings**<br>Not applicable | N/A | N/A | N/A |
| CNMT Loss 4 | **SG Secondary Side Release with P-to-S Leakage**<br>A. RUPTURED SG is also FAULTED | CNMT Loss C.3 | Ruptured S/G is also faulted outside of containment | None |

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| | outside of containment.<br>**OR**<br>B. a. Primary-to-Secondary leakrate greater than 10 gpm.<br>**AND**<br>b. UNISOLABLE steam release from affected SG to the environment. | CNMT Loss C.4 | Primary-to-secondary leakrate > 10 gpm<br>**AND**<br>Unisolable prolonged steam release from affected S/G to the environment | The NEI threshold has been divided into two Ginna thresholds to improve clarity.<br>Consistent with the generic developers guidance "Prolonged" as used here is in the context meaning that the release from the affected S/G within the time frame expected when implementing E-3 Steam Generator Tube Rupture. Cooldowns conducted to allow controlled isolation of the affected S/G per emergency procedures are not considered prolonged releases. The criterion for prolonged release is met if the objective of E-3 to isolate the affected S/G cannot be met. |
| CNMT Loss 5 | **Containment Isolation Failure or Bypass**<br>A. a. Failure of all valves in any one line to close.<br>**AND**<br>b. Direct downstream pathway to the environment exists after containment isolation signal. | CNMT Loss E.5 | Failure of **all** valves in **ANY one** line to close<br>**AND**<br>Direct downstream pathway to the environment exists after containment isolation signal | None |
| CNMT Loss 6 | **Containment Radiation Monitoring**<br>Not Applicable | N/A | N/A | N/A |
| CNMT Loss 7 | **Other (Site-Specific) Indications**<br>A. (Site-specific ) as applicable | N/A | None | Other site-specific indications of Containment loss have not been identified. |
| CNMT Loss 8 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier. | CNMT Loss F.6 | **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier | None |

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CNMT P-Loss 1 | **Critical Safety Function Status**<br>A. Containment-Red Entry Conditions Met. | CNMT P-Loss A.1 | **RED** path condition exists F-0.5 Containment | The NEI phrase "Containment-Red Entry Conditions Met" has been changed to "**RED** path condition exists F-0.5 Containment." Procedure F-0.5 is the containment CSFST.<br>The NEI phrase "Entry Conditions Met" has been changed to "condition exists" for consistency with terminology used by Ginna operators when using the EOPs. |
| CNMT P-Loss 2 | **Containment Pressure**<br>A. Containment pressure greater than *(site specific value)* and rising.<br>**OR**<br>B. Explosive mixture exists inside containment.<br>**OR**<br>C. a. Pressure greater than containment depressurization actuation setpoint.<br>**AND**<br>b. Less than one full train of depressurization equipment operating. | CNMT P-Loss C.4 | Containment pressure ≥ 60 psig and rising | The NEI threshold has been divided into three plant thresholds to improve clarity.<br>*This threshold is the containment design pressure and* is in excess of that expected from the design basis loss of coolant accident (LOCA). |
| | | CNMT P-Loss C.5 | Containment hydrogen concentration ≥ 4% | The NEI threshold has been divided into three Ginna thresholds to improve clarity.<br>Containment hydrogen concentration of 4% is the minimum concentration associated with an explosive mixture. |
| | | CNMT P-Loss C.6 | Containment pressure ≥ 28 psig and < two CRFC units and one CS pump operating per design | The NEI threshold has been divided into three Ginna thresholds to improve clarity.<br>The word "Containment" has been added to the plant threshold for clarification.<br>The Containment pressure setpoint (28 psig) is the Containment depressurization actuation setpoint.<br>The phrase "Less than one full train of depressurization equipment operating" has been replaced with "< two CRFC units and one CS pump operating per design." This represents the minimum design requirement for containment cooling. |

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CNMT P-Loss 3 | **Core Exit Thermocouple Readings**<br>A. a. Core exit thermocouples in excess of (site specific) ° F.<br>AND<br>b. Restoration procedures not effective within 15 minutes.<br>OR<br>B. a Core exit thermocouples in excess of (site-specific) F.<br>AND<br>b. Reactor vessel level below | CNMT P-Loss B.2 | Core Exit TCs **cannot** be restored < 1,200°F within 15 min. | The NEI threshold has been divided into two Ginna thresholds to improve clarity.<br>"TCs" is the Ginna equivalent of NEI "thermocouples."<br>The NEI phrase "in excess of (site specific) ° F AND Restoration procedures not effective..." has been replaced with "**cannot** be restored < 1,200°F...". The phrase "cannot be restored <" implies Core Exit TC readings have exceeded the threshold temperature and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful. Whether or not guidance is effective should be apparent within fifteen minutes. |

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| | (site specific level).<br>AND<br>c. Restoration procedures not effective within 15 minutes. | CNMT P-Loss<br>B.3 | Core Exit TCs ≥ 700°F<br>**AND**<br>RVLIS level **cannot** be restored > 52% [> 55% adverse CNMT] with no RCPs running within 15 min. | The NEI threshold has been divided into two Ginna thresholds to improve clarity.<br>"TCs" is the Ginna equivalent of NEI "thermocouples."<br>The NEI phrase "in excess of (site specific) ° F AND reactor vessel level below...AND Restoration procedures not effective..." has been replaced with "Core Exit TCs ≥ 700° **AND** RVLIS level **cannot** be restored > 52% [> 55% adverse CNMT] with no RCPs running...".<br>Core Exit TCs are a component of inadequate core cooling instrumentation and provide an indirect indication of fuel cladding temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 1 of EPIP-2-16, Core Damage Estimation . RCS superheat, as indicated by Core Exit TCs reading in excess of 700°F, signals the transition from a subcooled to a superheated regime. In a superheated regime, heat transfer mechanics are not as efficient as the subcooled condition and could lead to a rapid rise in cladding temperatures.<br>The phrase "cannot be restored >" implies RVLIS level readings have exceeded the threshold level and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful. Whether or not guidance is effective should be apparent within fifteen minutes. |
| CNMT P-Loss 4 | **SG Secondary Side Release with P-to-S Leakage**<br>Not applicable | N/A | N/A | N/A |

| NEI FPB# | NEI IC Wording | Ginna FPB #(s) | Ginna FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CNMT P-Loss 5 | **Containment Isolation Failure or Bypass**<br>Not Applicable | N/A | N/A | N/A |
| CNMT P-Loss 6 | **Containment Radiation Monitoring**<br>A. Containment radiation monitor reading greater than (site specific value). | CNMT P-Loss D.7 | Containment radiation monitor R-29/R-30 reading > 1.0E+03 R/hr | A reading on containment radiation monitors greater than 1.0E+03 R/hr is indicative of significant fuel activity and thus considered a potential loss of Containment. |
| CNMT P-Loss 7 | **Other (Site-Specific) Indications**<br>A. (Site-specific) as applicable | N/A | None | Other site-specific indications of Containment potential loss have not been identified. |
| CNMT P-Loss 8 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier. | CNMT P-Loss F.8 | **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier | None |

# Category H

# Hazards and Other Conditions Affecting Plant Safety

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU1 | Natural or destructive phenomena affecting the PROTECTED AREA.<br>MODE: All | HU1 | Natural or destructive phenomena affecting the Protected Area<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Seismic event identified by **ANY** 2 of the following:<br>• Seismic event confirmed by (site specific indication or method)<br>• Earthquake felt in plant<br>• National Earthquake Center | HU1.1 | Seismic event identified by **ANY two** of the following:<br>• Red LED event indicator on Kinematrics ETNA Digital Recorder indicates seismic event detected<br>• Earthquake felt onsite<br>• National Earthquake Information Center (Note 6)<br>Note 6: 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 Ginna Nuclear Power Plant. Provide the analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude**. | Red LED event indicator on Kinemetrics ETNA Digital Recorder provides the site specific indication or method of detecting a seismic event.<br>The NEI phrase "felt in plant" has been changed to "felt onsite" for consistency with the NEI basis which states: "the vibratory ground motion is felt at the nuclear plant site."<br>Note 6 provides guidance for contacting the NEIC and verifying seismic activity near the Ginna site. |
| 2 | Tornado striking within PROTECTED AREA boundary or high winds greater than (site specific mph). | HU1.2 | Tornado striking within Protected Area boundary<br>**OR**<br>Sustained high winds > 75 mph | The wind speed of 75 mph is the sustained design wind speed for Class 1 safe shutdown structures. |

| | | | | |
|---|---|---|---|---|
| 3 | Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in **ANY** of the following areas:<br>(site specific area list) | HU1.3 | Internal flooding that has the potential to affect **ANY** safety-related structure, system, or component required by Technical Specifications for the current operating mode in **ANY** Table H-1 area | The NEI phrase "safety related equipment" has been changed to "**ANY** safety-related structure, system, or component ..." for clarification<br>Ginna areas containing safety related equipment are specified in Table H-1. This change implements EAL FAQ #44. |
| 4 | Turbine failure resulting in casing penetration or damage to turbine or generator seals. | HU1.4 | Turbine failure resulting in casing penetration or damage to turbine or generator seals | None |
| 5 | (Site specific occurrences affecting the PROTECTED AREA). | HU1.5 | Deer Creek flooding over entrance road bridge hand rail<br>**OR**<br>Lake level > 252 ft<br>**OR**<br>Screen House Suction Bay water level < 17 ft or < 15.5 ft by manual level measurement | Flooding in Deer Creek above the plant entrance handrails will ultimately result in water accumulation in the Turbine Building and Screenhouse. This may preclude emergency response personnel access and egress.<br>A lake level of 252 ft has been selected for this threshold to be anticipatory of exceeding design flood levels and is the level at which flood control actions are procedurally taken.<br>When Screenhouse Suction Bay water level drops to 17.0 ft indicated (this corresponds to a level of 15.5 ft measured manually) increased Control Room monitoring is initiated. This level has been selected for this threshold to be anticipatory of a potential loss of service water system pump suction at 16.0 ft. |

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU2 | FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA.<br>MODE: All | HU2 | Fire within the Protected Area **not** extinguished within 15 min. of detection or explosion within the Protected Area<br>MODE: All | |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | FIRE not extinguished within 15 minutes of control room notification or verification of a control room FIRE alarm in **ANY** of the following areas:<br>(site specific area list) | HU2.1 | Fire **not** extinguished within 15 min. of Control Room notification or verification of a Control Room fire alarm in **ANY** Table H-1 area or Turbine Building (Note 4)<br>Note 4: The ED 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. | The NEI phrase "of the following areas...(site specific area list)" has been changed to "in **ANY** Table H-1 area." The areas listed in Table H-1 are areas containing functions and systems required for safe shutdown. This change implements EAL FAQ #44.<br>The phrase "or Turbine Building" has been added to the Ginna EAL to identify an area immediately adjacent to a Table H-1 area that could be susceptible to fires.<br>Note 4 has been added consistent with other NEI based EALs that include the 15 min. transitory condition exclusion.<br>The third paragraph of the NEI basis has been edited to clarify the significance of the 15-minute duration. If the alarm cannot be verified by redundant Control Room or nearby Fire Panel indications, notification from the field that a fire exists starts the concurrent 15-minute classification and fire suppression clocks. This change is consistent with the manner in which the Control Room and Fire Brigade leaders verify fires. This change is necessary to avoid declaring Unusual Event emergencies for spurious alarms that, due to the sensor location, cannot be verified within 15 minutes of receipt of the alarm. **This is a deviation from NEI 99-01 Revision 5.** |
| 2 | EXPLOSION within the | HU2.2 | Explosion within the Protected | None |

| | | | | |
|---|---|---|---|---|
| | PROTECTED AREA. | | Area | |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU3 | Release of toxic, corrosive, asphyxiant, or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS.<br>MODE: All | HU3 | Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS. | HU3.1 | Release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations | Revised wording to be consistent with the first sentence of the generic bases and consistent with the IC. |
| 2 | Report by local, county or state officials for evacuation or sheltering of site personnel based on an off-site event. | HU3.2 | Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event | Reworded EAL for readability. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU4 | Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.<br>MODE: All | HU4 | Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the (site specific security shift supervision). | HU4.1 | A security condition that does **not** involve a hostile action as reported by Security Shift Supervision<br>**OR**<br>A credible site-specific security threat notification<br>**OR**<br>A validated notification from NRC providing information of an aircraft threat | The NEI Example EALs have been combined in one plant EAL for simplification.<br>"Security Shift Supervision" is the site-specific security supervision that are qualified and trained to confirm that a security event is occurring or has occurred. |
| 2 | A credible site specific security threat notification. | | | |
| 3 | A validated notification from NRC providing information of an aircraft threat. | | | |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU5 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE.<br>MODE: All | HU6 | Other conditions existing that in the judgment of the Emergency Director warrant declaration of a UE<br>MODE: All | The NEI acronym "NOUE" has been implemented as "UE" for simplification.<br>Replaced the word "which" with "that" for proper grammar. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs. | HU6.1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant **OR** indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs | None |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA1 | Natural or destructive phenomena affecting VITAL AREAS<br>MODE: All | HA1 | Natural or destructive phenomena affecting Vital Areas<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. Seismic event greater than Operating Basis Earthquake (OBE) as indicated by (site specific seismic instrumentation) reading (site specific OBE limit).<br>**AND**<br>b. Earthquake confirmed by **ANY** of the following:<br>• Earthquake felt in plant<br>• National Earthquake Center<br>• Control Room indication of degraded performance of systems required for the safe shutdown of the plant. | HA1.1 | **EITHER**:<br>Confirmation of an earthquake of an intensity > 0.08 g per ER-SC.4 Earthquake Emergency Plan<br>**OR**<br>Control Room indication of degraded performance of **ANY** safety-related structure, system, or component<br>**AND**<br>Earthquake confirmed by **EITHER**:<br>Earthquake felt onsite<br>**OR**<br>National Earthquake Information Center (Note 6)<br>Note 6: 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 Ginna Nuclear Power Plant. Provide the | The site-specific instrumentation used to indicate a seismic event > OBE cannot be analyzed in a timely manner and requires analysis of the Strong Motion Accelerograph by site I&C and engineering per procedure ER-SC.4 Earthquake Emergency Plan. To allow for timely classification under this threshold, actual indication of degraded performance of safety-related structures systems or component has been included as a primary indicator of exceeding the OBE threshold.<br>The NEI phrase "felt in plant" has been changed to "felt onsite" for consistency with the NEI basis which states: "the vibratory ground motion is felt at the nuclear plant site."<br>Note 6 provides guidance for contacting the NEIC for confirmation of seismic activity in the vicinity of Ginna. |

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| | | | analyst with the following Ginna coordinates: **43° 16.7' north latitude, 77° 18.7' west longitude** | |
| 2 | Tornado striking or high winds greater than (site specific mph) resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems:<br><br>(site specific structure list) | HA1.2 | Tornado striking or sustained high winds > 75 mph resulting in **EITHER**:<br>Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area<br>**OR**<br>Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area | The wind speed of 75 mph is the sustained design wind speed for Class 1 safe shutdown structures.<br>The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the tornado/high wind could be presented in list format.<br>The NEI phrase "**ANY** of the following structures containing safety systems or components" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" to be consistent with the definition of visible damage and related HA1 EAL thresholds. This also permits presentation of the site specific list in a table.<br>Table H-1 provides the list of structures containing safety systems or components. This change implements EAL FAQ #44.<br>The NEI phrase "those safety systems" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" for clarification. |
| 3 | Internal flooding in **ANY** of the following areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment OR control room indication of degraded performance of those safety systems:<br>(site specific area list) | HA1.3 | Internal flooding in **ANY** Table H-1 area resulting in **EITHER**:<br>An electrical shock hazard that precludes access to operate or monitor **ANY** safety-related structure, system, or component within **ANY** Table H-1 area<br>**OR**<br>Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area | Ginna areas containing safety related equipment are specified in Table H-1. This change implements EAL FAQ #44.<br>The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the flooding could be presented in list format.<br>The NEI phrase "those safety systems" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" for clarification. |

| | | | | |
|---|---|---|---|---|
| 4 | Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems:<br>(site specific structure list) | HA1.4 | Turbine failure-generated projectiles resulting in **EITHER**:<br>Visible damage to or penetration of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area<br>**OR**<br>Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area | The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the turbine failure-generated projectiles could be presented in list format.<br>The NEI phrase "**ANY** of the following structures containing safety systems or components" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" to be consistent with the definition of visible damage and related HA1 EAL thresholds. This also permits presentation of the site specific list in a table.<br>Table H-1 provides the list of areas/structures containing safety systems or components. This change implements EAL FAQ #44.<br>The NEI phrase "those safety systems" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" for clarification. |
| 5 | Vehicle crash resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems:<br>(site specific structure list) | HA1.6 | Vehicle crash resulting in **EITHER**:<br>Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area<br>**OR**<br>Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area | The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the vehicle crash could be presented in list format.<br>The NEI phrase "**ANY** of the following structures containing safety systems or components" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" to be consistent with the definition of visible damage and related HA1 EAL thresholds. This also permits presentation of the site specific list in a table.<br>Table H-1 provides the list of areas/structures containing safety systems or components. This change implements EAL FAQ #44.<br>The NEI phrase "those safety systems" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" for clarification. |
| 6 | (Site specific occurrences) resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety | HA1.5 | Lake level > 253 ft<br>**OR**<br>Screen House Suction Bay water level < 16 ft or < 14.5 ft by manual level measurement | Lake water level > 253.28 ft msl corresponds to plant design levels (ref. 1). A lake level of 253 ft has been selected for this threshold to be indicative of exceeding design flood levels.<br>If indicated service water pump bay level drops below 16 ft (this corresponds to a lake level of 14.5 ft measured manually) the service water pumps are declared inoperable. This level has been |

| | | | | |
|---|---|---|---|---|
| | systems: | | | selected for this threshold to be indicative of a loss of service water system pump suction. |

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA2 | FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown<br>MODE: All | HA2 | Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | FIRE or EXPLOSION resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems:<br>(site specific structure list) | HA2.1 | Fire or explosion resulting in **EITHER**:<br>Visible damage to **ANY** safety-related structure, system, or component within **ANY** Table H-1 area<br>**OR**<br>Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area | The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the fire/explosion could be presented in list format.<br>The NEI phrase "**ANY** of the following structures containing safety systems or components" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" so that the site specific list could be presented in a table.<br>Table H-1 provides the list of areas/structures containing safety systems or components. This change implements EAL FAQ #44.<br>The NEI phrase "those safety systems" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" for clarification. |

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Containment Building<br>• Auxiliary Building<br>• Control Building<br>• Intermediate Building<br>• Emergency Diesel Building(s)<br>• SAFW Building<br>• Screenhouse<br>• Cable Tunnel<br>• Battery Rooms |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA3 | Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor.<br>MODE: All | HA3 | Access to a Vital Area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shut down the reactor<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.<br>1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor. | HA3.1 | Access to **ANY** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of **ANY** safety-related structure, system, or component (Note 5)<br>Note 5: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should **not** be declared as it will have **no** adverse impact on the ability of the plant to safely operate or safely shut down beyond that already allowed by Technical Specifications at the time of the event. | Ginna vital areas are specified in Table H-1. This change implements EAL FAQ #44.<br>The NEI phrase "systems required to maintain safe operations or safely shutdown the reactor" has been changed to "**ANY** safety-related structure, system, or component" for consistency with subcategory H1 and H2 Alert EALs. Safety-related structures, systems, and components are systems that are required to maintain safe operations or safely shut down the reactor.<br>Reference to the NEI note is included in the EAL wording "(Note 5)." Numbering the note facilitates referencing in the EAL matrix.<br>The NEI phrase "this EAL" has been changed to "EAL HA3.1" for clarification. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA4 | HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.<br>MODE: All | HA4 | Hostile action within the Owner Controlled Area or airborne attack threat.<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site specific security shift supervision). | HA4.1 | A hostile action is occurring or has occurred within the Owner Controlled Area as reported by Security Shift Supervision<br>**OR**<br>A validated notification from NRC of an airliner attack threat within 30 min. of the site | The NEI Example EALs have been combined in one plant EAL for simplification.<br>"Security Shift Supervision" is the site-specific security supervision that are qualified and trained to confirm that a security event is occurring or has occurred. |
| 2 | A validated notification from NRC of an airliner attack threat within 30 minutes of the site. | | | |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA5 | Control Room Evacuation Has Been Initiated<br>MODE: All | HA5 | Control Room evacuation has been initiated<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | (Site-specific procedure) requires control room evacuation. | HA5.1 | Control Room evacuation has been initiated | AP-CR.1 Control Room Inaccessibility provides specific instructions for evacuating the Control Room and establishing plant control in alternate locations. The IC wording has been utilized since the intent is to classify the Alert based on Control Room evacuation, regardless whether the associated procedure has been entered or executed. This change implements EAL FAQ #28. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA6 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.<br>MODE: All | HA6 | Other conditions exist that in the judgment of the Emergency Director warrant declaration of an Alert<br>MODE: All | Replaced the word "which" with "that" for proper grammar. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels. | HA6.1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant **OR** a security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) | EPA PAG values have been added for clarification. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HS2 | Control room evacuation has been initiated and plant control cannot be established.<br>MODE: All | HS5 | Control Room evacuation has been initiated and plant control **cannot** be established<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. Control room evacuation has been initiated.<br>**AND**<br>b. Control of the plant cannot be established within (site specific minutes). | HS5.1 | Control Room evacuation has been initiated<br>**AND**<br>Control of the plant **cannot** be established within 30 min. | AP-CR.1 Control Room Inaccessibility provides specific instructions for evacuating the Control Room and establishing plant control in alternate locations.<br>An analysis was performed as part of the Fire Protection Program to determine how quickly control must be re-established at Ginna without core uncovery or damage. There are 5 time-critical actions which must be accomplished to enable established performance goals to be met. In evaluating a reasonable timeline for completion of tasks required in the ER-FIRE procedures to restore charging, it was estimated that restoration should be completed in less than 30 minutes. This is consistent with information obtained during operator walk-throughs of the ER-FIRE procedures which consistently indicated restoration in 17 to 24 minutes. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HS3 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.<br>MODE: All | HS6 | Other conditions existing that in the judgment of the Emergency Director warrant declaration of a Site Area Emergency<br>MODE: All | Replaced the word "which" with "that" for proper grammar |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary. | HS6.1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public **OR** hostile action that results in intentional damage or malicious acts: (1) toward site personnel or equipment that could lead to the likely failure of, or (2) that prevent effective access to, equipment needed for the protection of the public. **Any** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) beyond the site boundary | EPA PAG values have been added for clarification. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HS4 | HOSTILE ACTION within the PROTECTED AREA<br>MODE: All | HS4 | Hostile action within the Protected Area<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site security shift supervision) | HS4.1 | A hostile action is occurring or has occurred within the Protected Area as reported by Security Shift Supervision | The "Security Shift Supervisor" is the title of the site-specific security shift supervision. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HG1 | HOSTILE ACTION resulting in loss of physical control of the facility.<br>MODE: All | HG4 | Hostile action resulting in loss of physical control of the facility<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. | HG4.1 | A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions | None |
| 2 | A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool. | HG4.2 | A hostile action has caused failure of Spent Fuel Cooling systems<br>**AND**<br>Imminent fuel damage is likely | The logic term "**AND**" has been added to the threshold for format consistency.<br>The NEI phrase "for a freshly off-loaded reactor core in pool" has been deleted because any imminent fuel damage caused by hostile action warrants a GE declaration even if it is not from a freshly off-loaded core in pool. This change implements EAL FAQ # 29. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HG2 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.<br>MODE: All | HG6 | Other conditions exist that in the judgment of the Emergency Director warrant declaration of a General Emergency<br>MODE: All | Replaced the word "which" with "that" for proper grammar |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels off-site for more than the immediate site area. | HG6.1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity **OR** hostile action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) offsite for more than the immediate site area | EPA PAG values have been added for clarification. |

# Category S

# System Malfunction

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU1 | Loss of all Off-site AC power to emergency busses for 15 minutes or longer.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU1 | Loss of **all** offsite AC power to 480V safeguards buses for ≥ 15 min.<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | "480V safeguards buses" is the Ginna specific terminology for "emergency busses." |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note**: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>Loss of all off-site AC power to (site specific emergency busses) for 15 minutes or longer. | SU1.1 | Loss of **all** offsite AC power, Table S-1, to 480V safeguards buses for ≥ 15 min. (Note 4)<br>Note 4: The ED 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. | Table S-1 provides a list of onsite and offsite AC power supplies.<br>480V safeguards buses are the Ginna emergency buses in hot conditions.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| Table S-1 AC Power Sources | |
|---|---|
| **Onsite** | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| **Offsite** | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU2 | Inability to reach required shutdown within Technical Specification limits.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU4 | Inability to reach required shutdown within Technical Specification limits<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time. | SU4.1 | Plant is **not** brought to required operating mode within Technical Specifications LCO required action completion time | "... required action completion time" is the Ginna Technical Specification terminology for "... Action Statement Time." |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU3 | UNPLANNED loss of safety system annunciation or indication in the control room for 15 minutes or longer.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU5 | Unplanned loss of safety system annunciation or indication in the Control Room for ≥ 15 min.<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>UNPLANNED Loss of greater than approximately 75% of the following for 15 minutes or longer:<br>a. (Site specific control room safety system annunciation)<br>**OR**<br>b. (Site specific control room safety system indication) | SU5.1 | Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)<br>Note 4: The ED 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. | The NEI phrase "…greater than approximately 75% of the following …a…b…" has been changed to "…loss of 6 or more annunciator panels, Table S-2 or >75% of MCB indications," for clarity. Table S-2 lists the safety-related annunciation panels. Loss of 6 annunciator panels would be a loss of 75% of the safety related annunciators.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| Table S-2 Vital Control Room Panels | | | | | | | |
|---|---|---|---|---|---|---|---|
| A | AA | B | C | D | E | F | G |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU4 | Fuel Clad Degradation<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU7 | Fuel clad degradation<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | (Site specific radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits.) | SU7.2 | Valid Letdown Monitor (R-9) reading ≥ 4800 mRad/hr | The Letdown Line Monitor (R-9) gross radiation channel continuously monitors the activity in a sample drawn from the RCS (NaOH tank room) and actuates an alarm in the Control Room if a predetermined activity level is reached. 4800 mRad/hr corresponds to the transient TS coolant activity limit of 60 µCi/gm (3.4.16). |
| 2 | (Site specific coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.) | SU7.1 | RCS specific activity > 60 µCi/gm dose equivalent I-131 | Consistent with the generic NEI 99-01 Rev. 5 bases, the Technical Specification threshold limit value selected is 60 µCi/gm dose equivalent I-131. This limit accommodates iodine spike phenomenon that may occur following changes in thermal power and during reactor startup and shutdown. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU5 | RCS Leakage<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU8 | RCS leakage<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Unidentified or pressure boundary leakage greater than 10 gpm. | SU8.1 | Unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 4)<br>**OR**<br>Identified leakage > 25 gpm for ≥ 15 min. (Note 4)<br>Note 4: The ED 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. | SU8.1 implements Example EALs #1 and #2. These were combined for improved usability.<br>The phrase "for ≥ 15 min. (Note 4)" has been added to the Ginna EAL to allow mitigation by operating procedures prior to declaration. **This is a deviation from NEI 99-01 Revision 5.** |
| 2 | Identified leakage greater than 25 gpm, | | | |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU6 | Loss of all On-site or Off-site communications capabilities.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU6 | Loss of **all** onsite or offsite communications capabilities<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss of all of the following on-site communication methods affecting the ability to perform routine operations.<br>(site specific list of communications methods) | SU6.1 | Loss of **all** Table S-4 onsite (internal) communication methods affecting the ability to perform routine operations<br>**OR**<br>Loss of **all** Table S-4 offsite (external) communication methods affecting the ability to perform offsite notifications | SU6.1 implements Example EALs #1 and #2. These were combined for improved usability.<br>The NEI example EALs specify site-specific lists of onsite and offsite communications methods. The Ginna EAL lists these methods in Table S-4 because the number of communications methods is too long to include within the text of the EAL.<br>The adjectives "(internal)" and "(external)" have been added to the Ginna EAL for clarification. The terms "onsite/offsite" could be interpreted as the location in which the communication originates instead of the location to which communication is directed. |
| 2 | Loss of all of the following off-site communication methods affecting the ability to perform offsite notifications.<br>(site specific list of communications methods) | | | |

| Table S-4 Communications Systems | | |
|---|---|---|
| **System** | **Onsite** (internal) | **Offsite** (external) |
| Commercial phone system | X | X |
| Direct Dial "POTS" Lines (Blue Phones) | X | X |
| Plant Page Party system | X | |
| Radios/Walkie Talkies | X | |
| FTS 2001 telephone system (ENS, HPN) | | X |
| Control Room Hard Wired Satellite Phone | | X |
| Control Room Emergency Cell Phone | | X |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU8 | Inadvertent Criticality.<br>MODE: Hot Standby, Hot Shutdown | SU3 | Inadvertent criticality<br>MODE: 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | UNPLANNED sustained positive period observed on nuclear instrumentation. [*BWR*] | N/A | N/A | NEI Example EAL #1 has not been implemented because it applies only to BWR plants. Ginna is a PWR. PWRs are not equipped with period meters. |
| 1 | UNPLANNED sustained positive startup rate observed on nuclear instrumentation. [*PWR*] | SU3.1 | An unplanned sustained positive startup rate observed on nuclear instrumentation | None |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SA2 | Automatic Scram (Trip) fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor.<br>MODE: Power Operation, Startup | SA3 | Automatic trip failed to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor<br>MODE: 1 - Power Operation | The term "scram" was replaced with "trip" consistent with PWR terminology.<br>The NEI term "fails" has been changed to "failed" for consistency with the example EAL wording. This change implements EAL FAQ #31.<br>The Startup mode has been deleted from the Ginna EAL. Ginna Technical Specifications definition of Startup mode is $K_{eff} \geq 0.99$ and rated thermal power ≤ 5%. It is not possible to be in Startup mode with reactor power above 5%. Since the definition of reactor shutdown is reactor power less than or equal to 5% (in accordance with the NEI 99-01 basis for this EAL), this EAL would never be applicable in Startup mode. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. An automatic scram (trip) failed to shutdown the reactor.<br>**AND**<br>b. Manual actions taken at the reactor control console successfully shutdown the reactor as indicated by (site specific indications of plant shutdown). | SA3.1 | An automatic trip failed to shut down the reactor<br>**AND**<br>Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power ≤ 5% | The term "scram" was replaced with "trip" consistent with PWR terminology.<br>The power range indication above 5% is greater than the decay heat which the shutdown systems (Auxiliary Feed Water and Atmospheric Dump Valves) were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SA4 | UNPLANNED Loss of safety system annunciation or indication in the control room with EITHER (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory indicators unavailable.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SA5 | Unplanned loss of safety system annunciation or indication in the Control Room with either (1) a significant transient in progress, or (2) compensatory indicators are unavailable<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>a. UNPLANNED loss of greater than approximately 75% of the following for 15 minutes or longer:<br>• (Site specific control room safety system annunciation)<br>**OR**<br>• (Site specific control room safety system indication)<br>b. **EITHER** of the following:<br>• A SIGNIFICANT TRANSIENT is in progress. | SA5.1 | Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)<br>**AND EITHER**:<br>A significant transient is in progress, Table S-3<br>**OR**<br>Compensatory indications are unavailable (PPCS)<br>Note 4: The ED 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. | The NEI phrase "...greater than approximately 75% of the following ...a...b..." has been changed to "...loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications," for clarity. Table S-2 lists the safety-related annunciation panels. Loss of 6 annunciator panels would be a loss of 75% of the safety related annunciators.<br>Table S-3 provides the list of events that constitute a "significant transient" as specified in the NEI 99-01 Section 5.4 definition of significant transient.<br>The Ginna compensatory indications are provided by PPCS.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| | | | | |
|---|---|---|---|---|
| | • Compensatory indications are unavailable. | | | |

| Table S-3 Significant Transients |
|---|
| • Automatic turbine runback > 25% thermal power |
| • Electric load rejection > 25% full electrical load |
| • Reactor trip |
| • Safety Injection activation |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SA5 | AC power capability to emergency busses reduced to a single power source for 15 minutes or longer such that any additional single failure would result in station blackout.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SA1 | AC power capability to 480V safeguards buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 480V safeguards bus power<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | "480V safeguards buses" is the Ginna specific terminology for "emergency busses."<br>The phrase "... **ANY** additional single failure would result in station blackout." was replaced with "... **ANY** additional single failure would result in a complete loss of all 480V safeguards bus power." This is consistent with the intent that classification be based on a loss of AC power to emergency buses. A Station Blackout involves a loss of all AC power, not just emergency bus power. This change implements EAL FAQ #36. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>a. AC power capability to (site-specific emergency busses) reduced to a single power source for 15 minutes or longer.<br>b. Any additional single power source failure will result in station blackout. | SA1.1 | AC power capability to 480V safeguards buses reduced to a single power source, Table S-1, for ≥ 15 min. (Note 4)<br>**AND**<br>**Any** additional single power source failure will result in a complete loss of all 480V safeguards bus power<br>Note 4: The ED 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. | Table S-1 provides a list of onsite and offsite AC power supplies.<br>480V safeguards buses are the Ginna emergency buses in hot conditions.<br>The NEI phrase "...station blackout" has been replaced with "... complete loss of all 480V safeguards bus power" as this describes the intended condition for Ginna. This change implements EAL FAQ #36.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| Table S-1 AC Power Sources | |
|---|---|
| Onsite | • EDG 1A (Bus 14)<br>• EDG 1B (Bus 16) |
| Offsite | • Station Auxiliary Transformer 12A<br>• Station Auxiliary Transformer 12B<br>• Unit Auxiliary Transformer 11 backfeed (if currently established) |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SS1 | Loss of all Off-site and all On-Site AC power to emergency busses for 15 minutes or longer.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SS1 | Loss of **all** offsite and **all** onsite AC power to 480V safeguards buses for ≥ 15 min.<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | "480V safeguards buses" is the Ginna specific terminology for "emergency busses." |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>Loss of all Off-Site and all On-Site AC power to (site specific emergency busses) for 15 minutes or longer. | SS1.1 | Loss of **all** offsite and **all** onsite AC power, Table S-1, to 480V safeguards buses for ≥ 15 min. (Note 4)<br>Note 4: The ED 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. | Table S-1 provides a list of onsite and offsite AC power supplies.<br>480V safeguards buses are the Ginna emergency buses in hot conditions.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SS2 | Automatic Scram (Trip) fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting down the reactor.<br>MODE: Power Operation, Startup | SS3 | Automatic trip and manual actions taken from the reactor control console failed to shut down the reactor<br>1 - Power Operation | The term "scram" was replaced with "trip" consistent with PWR terminology.<br>The NEI phrase "fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting" has been changed to "and manual actions taken from the reactor control console failed to shut" for clarification. This change implements EAL FAQ #31.<br>The Startup mode has been deleted from the Ginna EAL. Ginna Technical Specifications definition of Startup mode is $K_{eff} \geq 0.99$ and rated thermal power ≤ 5%. It is not possible to be in Startup mode with reactor power above 5%. Since the definition of reactor shutdown is reactor power less than or equal to 5% (in accordance with the NEI 99-01 basis for this EAL), this EAL would never be applicable in Startup mode. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. An automatic scram (trip) failed to shutdown the reactor.<br>**AND**<br>b. Manual actions taken at the reactor control console do not shutdown the reactor as indicated by (site specific indications of reactor not shutdown). | SS3.1 | An automatic trip failed to shut down the reactor as indicated by reactor power > 5%<br>**AND**<br>Manual actions taken at the reactor control console failed to shut down the reactor as indicated by reactor power > 5% | The term "scram" was replaced with "trip" consistent with PWR terminology.<br>The phrase "as indicated by reactor power > 5%" has been added to the first contingent and the NEI phrase "do not shutdown the reactor" has been changed to "failed to shut down the reactor" in the second contingent for clarification and consistency of wording. This change implements EAL FAQ #31.<br>The power range indication above 5% is greater than the decay heat which the shutdown systems (Auxiliary Feed Water and Atmospheric Dump Valves) were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SS3 | Loss of all vital DC power for 15 minutes or longer.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SS2 | Loss of **all** vital DC power for ≥ 15 min.<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>Less than (site specific bus voltage indication) on all (site specific Vital DC busses) for 15 minutes or longer. | SS2.1 | < 108 VDC on **both** 125 VDC buses 1A and 1B for ≥ 15 min. (Note 4)<br>Note 4: The ED 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. | "108 VDC" is the site-specific bus voltage indication.<br>125 VDC buses 1A and 1B are the Ginna vital DC buses.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SS6 | Inability to monitor a SIGNIFICANT TRANSIENT in progress.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SS5 | Inability to monitor a significant transient in progress<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | None |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>a. Loss of greater than approximately 75% of the following for 15 minutes or longer:<br>• (Site specific control room safety system annunciation)<br>**OR**<br>• (Site specific control room safety system indication)<br>**AND**<br>b. A SIGNIFICANT TRANSIENT is in progress.<br>**AND** | SS5.1 | Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)<br>**AND**<br>A significant transient is in progress, Table S-3<br>**AND**<br>Compensatory indications are unavailable (PPCS)<br>Note 4: The ED 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. | The NEI phrase "...greater than approximately 75% of the following ...a...b..." has been changed to "...loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications," for clarity. Table S-2 lists the safety-related annunciation panels. Loss of 6 annunciator panels would be a loss of 75% of the safety related annunciators.<br>Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.<br>Table S-3 provides the list of events that constitute a "significant transient" as specified in the NEI 99-01 Section 5.4 definition of significant transient.<br>The Ginna compensatory indications are provided by PPCS. |

| | | | | |
|---|---|---|---|---|
| | c. Compensatory indications are unavailable. | | | |

| Table S-3 Significant Transients |
|---|
| • Automatic turbine runback > 25% thermal power<br>• Electric load rejection > 25% full electrical load<br>• Reactor trip<br>• Safety Injection activation |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SG1 | Prolonged loss of all Off-site and all On-Site AC power to emergency busses.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SG1 | Prolonged loss of **all** offsite and **all** onsite AC power to 480V safeguards buses<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Hot Standby | "480V safeguards buses" is the Ginna specific terminology for "emergency busses." |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. Loss of all off-site and all on-site AC power to (site specific emergency busses).<br>**AND**<br>b. **EITHER** of the following:<br>• Restoration of at least one emergency bus in less than (site specific hours) is not likely.<br>• (Site specific indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.) | SG1.1 | Loss of **all** offsite and **all** onsite AC power, Table S-1, to 480V safeguards buses<br>**AND EITHER:**<br>Restoration of at least one 480V safeguards bus within 4 hours is **not** likely<br>**OR**<br>**ORANGE** or **RED** path condition exists F-0.2 Core Cooling | 480V safeguards buses are the Ginna emergency buses in hot conditions.<br>The NEI phrase "...of the following: ..." has been deleted. It is evident from the subsequent paragraphs and indentation applied to the Ginna EAL that they follow the previous paragraph.<br>4 are the "(site-specific)" hours for station blackout coping. The four-hour interval to restore AC power is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155.<br>The NEI phrase "...(Site-Specific) Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring" has been replaced with "**ORANGE** or **RED** path condition exists F-0.2 Core Cooling" for clarification. This threshold represents the NEI conditions consistent with the corresponding fission product barrier Fuel Clad Loss and Potential Loss thresholds. |

| NEI IC# | NEI IC Wording | Ginna IC#(s) | Ginna IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SG2 | Automatic Scram (Trip) and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists.<br>MODE: Power Operation, Startup | SG3 | Automatic trip and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists<br>MODE: 1 - Power Operation | The term "scram" was replaced with "trip" consistent with PWR terminology.<br>The Startup mode has been deleted from the Ginna EAL. Ginna Technical Specifications definition of Startup mode is $K_{eff} \geq 0.99$ and rated thermal power ≤ 5%. It is not possible to be in Startup mode with reactor power above 5%. Since the definition of reactor shutdown is reactor power less than or equal to 5% (in accordance with the NEI 99-01 basis for this EAL), this EAL would never be applicable in Startup mode. |

| NEI Ex. EAL # | NEI Example EAL Wording | Ginna EAL # | Ginna EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. An automatic scram (trip) failed to shutdown the reactor.<br>**AND**<br>b. All manual actions do not shutdown the reactor as indicated by (site specific indications of reactor not shutdown).<br>**AND**<br>c. **EITHER** of the following exist or have occurred due to continued power generation:<br>• (Site specific indication that core cooling is extremely challenged.)<br>(Site specific indication that heat removal is extremely challenged.) | SG3.1 | An automatic trip failed to shut down the reactor as indicated by reactor power > 5%<br>**AND**<br>**All** manual actions fail to shut down the reactor as indicated by reactor power > 5%<br>**AND EITHER** of the following exist or have occurred:<br>**RED** path condition exists F-0.2 Core Cooling<br>**OR**<br>**RED** path condition exists F-0.3 Heat Sink | The term "scram" was replaced with "trip" consistent with PWR terminology.<br>The phrase "as indicated by reactor power > 5%" has been added to the Ginna EAL for clarification. This change implements EAL FAQ #31.<br>The power range indication above 5% is greater than the decay heat which the shutdown systems (Auxiliary Feed Water and Atmospheric Dump Valves) were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage.<br>The NEI phrase "do not shutdown" has been changed to "fail to shut down" for consistency with the IC wording. This change implements EAL FAQ #31.<br>The NEI phrase "due to continued power generation" has been deleted because the extreme challenge to heat removal, equivalent to core cooling red, should not be constrained by requiring it to be caused by continued power generation. This change implements EAL FAQ #37.<br>The NEI example EAL specifies site-specific indication that core |

| | | | | |
|---|---|---|---|---|
| | | | | cooling is extremely challenged and site-specific indication that heat removal is extremely challenged. The Ginna EAL includes the specific combination of conditions (**RED** path condition exists F-0.2 Core Cooling or **RED** path condition exists F-0.3 Heat Sink) that indicates the core cooling or ultimate heat sink function is under extreme challenge and therefore a core melt sequence may exist and rapid degradation of the fuel cladding could begin. |

# ATTACHMENT (4)

# RADIATION MONITOR SUPPORTING CALCULATIONS

# CALCULATION COVER SHEET

| | | | | |
|---|---|---|---|---|
| **A. INITIATION** | | | | Page 1 of 37 |
| Site | ☐ CCNPP | ☐ NMP | ☒ REG | |
| Calculation No.: | **CALC-2011-0019** | | Revision No.: | 000 |
| Vendor Calculation (Check one): | ☐ Yes | ☒ No | | |
| Responsible Group: | SHANE R. GARDNER | | | |
| Responsible Engineer: | NUCLEAR ANALYSIS UNIT - FLEET NUCLEAR FUELS | | | |

| | | | |
|---|---|---|---|
| **B. CALCULATION** | | | |
| ENGINEERING DISCIPLINE: | ☐ Civil | ☐ Instr & Controls | ☒ Nuc Eng |
| | ☐ Electrical | ☐ Mechanical | ☐ Nuc Fuel Mgmt |
| | ☐ Other: | ☐ Reliability Eng | |
| Title: | NEI 99-01 TECHNICAL BASIS FOR THE GINNA R-9 LETDOWN LINE MONITOR EMERGENCY ACTION LEVELS (EALS) | | |
| Unit: | ☒ 1 | ☐ 2 | ☐ COMMON |
| Proprietary or Safeguards Calculation: | | ☐ YES | ☒ NO |
| Comments: | SUPPORTS NEI 99-01 UPGRADE | | |
| Vendor Calc No.: | N/A | REVISION NO.: | N/A |
| Vendor Name: | N/A | | |
| Safety Class (Check one): | ☐ SR | ☐ AUGMENTED QUALITY | ☒ NSR |

There are assumptions that require Verification during walkdown: N/A    TRACKING ID:

This calculation **SUPERSEDES**: EAL-TECH-BASES-R9-NUE

**C. REVIEW AND APPROVAL:**

| | | |
|---|---|---|
| Responsible Engineer: | S. R. GARDNER *Shane R. Gardner* | 12/19/2011 |
| | Printed Name and Signature | Date |

Is Design Verification Required? ☐ Yes ☒ No

If yes, Design Verification Form is ☐ Attached ☐ Filed with:

| | | |
|---|---|---|
| Independent Reviewer: | J. R. MASSARI *[signature]* | 12/19/2011 |
| | Printed Name and Signature | |
| Approval: | *Philip Wengloski* *[signature]* | *12/20/11* |
| | Printed Name and Signature | Date |

## REVISION SUMMARY AND LIST OF EFFECTIVE PAGES

Initial Issue – All pages Rev. 0.

## TABLE OF CONTENTS

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## LIST OF FIGURES

## 1 OBJECTIVE AND PURPOSE

The objective of this calculation is to evaluate the letdown line radiation monitor, R-9 (Reference 6.1), response during Emergency Action Level (EAL) conditions specified in NEI 99-01 Revision 5 (Reference 6.2). The calculation will provide the radiation levels at the monitor due to various degrees of fuel failure.

## 2 SCOPE OF CALCULATION

The R-9 monitor results correspond specifically to the technical basis guidance specified in the NEI 99-01 Rev. 5. Future changes in the EAL technical bases will require review of this calculation for continued applicability or revision.

## 3 CONCLUSIONS

The R-9 monitor EAL values based on NEI 99-01 Rev. 5 are shown in Table 3-1. These values correspond to a notification of Unusual Event (UE) on fuel cladding degradation (SU4, action level 1) and an Alert on exceeding the threshold for loss of the fuel cladding fission product barrier (2.A., Table 5-F-3). Note these results are only valid when letdown is in operation and the Alert value may be offscale for the R-9 monitor.

**TABLE 3-1 R-9 EAL VALUES BASED ON NEI 99-01 REV. 5**

| Fuel Failure | DE I-131 (uCi/gm) | Frac 100/EBAR | EAL | R-9 EAL Value (mrad/hr) |
|---|---|---|---|---|
| 1% | 60 | 1 | UE | 4,800 |
| 5% | 300 | 5 | Alert | 24,000 |

## 4 DESIGN INPUTS

Design input data are specified below. Key inputs are provided as indicated. Other data used may not be replicated specifically below, but are traceable to the source provided.

4.1. The design basis Reactor Coolant System (RCS) activity concentrations for 1% failed fuel are shown in Table 4-1 based on Westinghouse calculation CN-REA-04-34 (Reference 6.3).

4.2. Letdown line data is taken from Gilbert drawing C-381-357 (Reference 6.4). The geometry of the piping based on this drawing is shown in Figure 4-1 along with the approximate position of R-9. The pipe data from the drawing also used is as follows:

a) Outside diameter – 2.375"

b) Wall thickness – 0.145" (601R line spec)

4.3. Reactor coolant mass is 1.123E+08 grams (Reference 6.3).

4.4. Iodine gas gap fractions for 35,000 MWd/MTU fuel are taken from DA-NS-08-49 (Reference 6.5, Table 6.2) as: I-131, 0.0039; I-132, 0.0004; I-133, 0.0013; I-134, 0.0003; I-135, 0.0007.

4.5. Iodine dose conversion factors from ICRP 30 (Reference 6.6) are used to calculate the Dose Equivalent (DE) I-131 coolant activity concentration. This is consistent with CHI-PRI-EBAR (Reference 6.7) for TS 3.4.16 surveillance

4.6. Average energy per disintegration data from Attachments 1 and 2 from CHI-PRI-EBAR are used for E-bar calculations. For nuclides not specified in CHI-PRI-EBAR, ICRP 38 (Reference 6.8) data are used.

4.7. Normal RCS activities used are based on Westinghouse calculation CN-REA-04-57 (Reference 6.9).

4.8. Design basis core activities used are based on Westinghouse calculations CN-REA-04-32 (Reference 6.10) for EPU operation and CN-REA-02-33 (Reference 6.11) for pre-EPU operation.

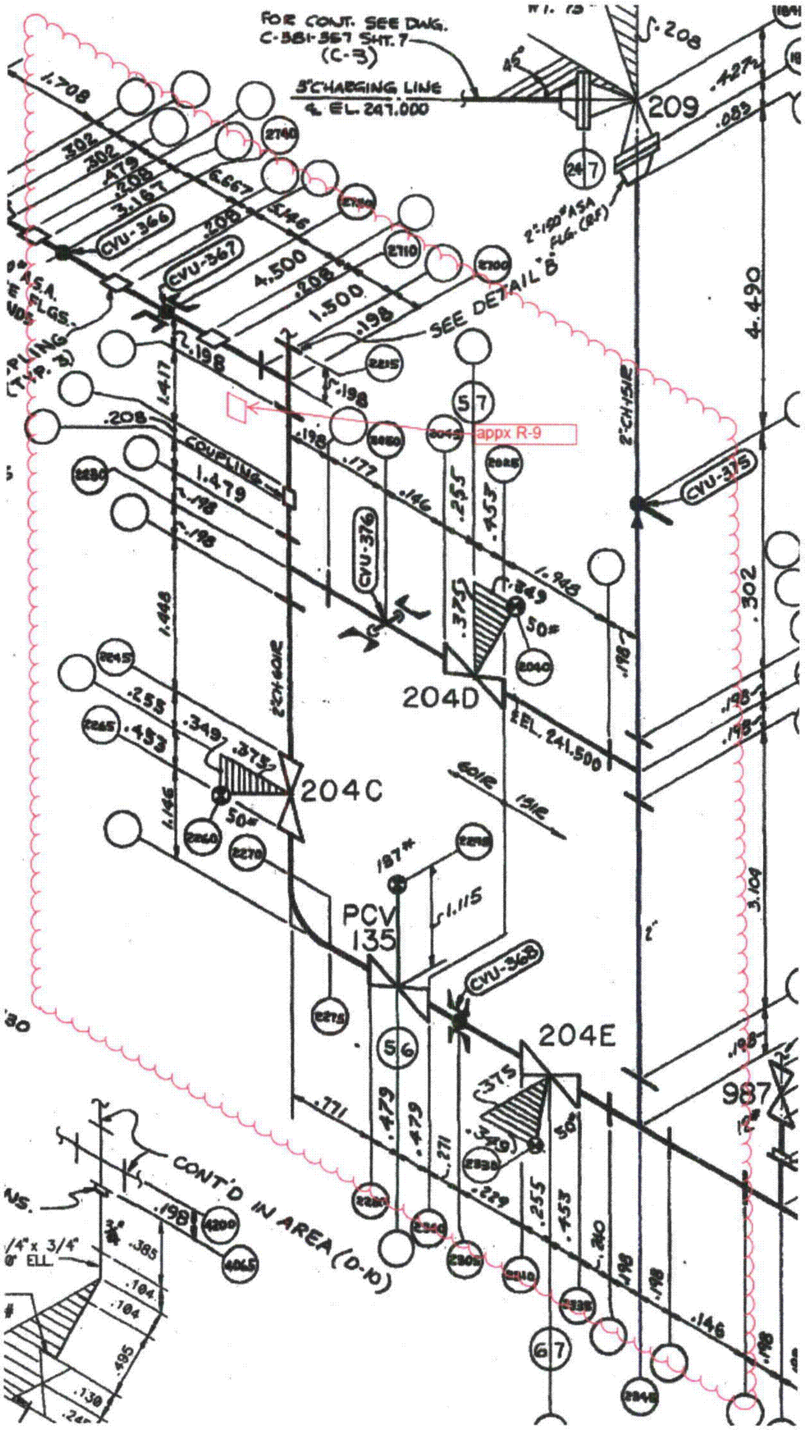
4.9. Flux-to-dose rate conversion factors from ANSI/ANS-6.1.1-1977 are used to calculate equivalent dose rates. These values are provided in the MCNP manual (Reference 6.12), Appendix H.

## TABLE 4-1 GINNA EPU DESIGN BASIS RCS ACTIVITY CONCENTRATONS

| Nuclide | Activity (uCi/gm) | Nuclide | Activity (uCi/gm) |
|---|---|---|---|
| Kr-83m | 4.74E-01 | Rb-88 | 4.40E+00 |
| Kr-85m | 1.93E+00 | Rb-89 | 2.00E-01 |
| Kr-85 | 8.21E+00 | Sr-89 | 4.56E-03 |
| Kr-87 | 1.24E+00 | Sr-90 | 2.33E-04 |
| Kr-88 | 3.60E+00 | Sr-91 | 6.00E-03 |
| Kr-89 | 1.00E-01 | Sr-92 | 1.32E-03 |
| Xe-131m | 3.54E+00 | Y-90 | 6.68E-05 |
| Xe-133m | 3.84E+00 | Y-91m | 3.26E-03 |
| Xe-133 | 2.71E+02 | Y-91 | 6.00E-04 |
| Xe-135m | 5.58E-01 | Y-92 | 1.16E-03 |
| Xe-135 | 9.49E+00 | Y-93 | 3.96E-04 |
| Xe-137 | 1.91E-01 | Zr-95 | 6.99E-04 |
| Xe-138 | 6.92E-01 | Nb-95 | 7.03E-04 |
| Br-83 | 1.00E-01 | Mo-99 | 8.38E-01 |
| Br-84 | 4.90E-02 | Tc-99m | 7.78E-01 |
| Br-85 | 5.70E-03 | Ru-103 | 6.11E-04 |
| | | Rh-103m | 6.14E-04 |
| I-129 | 6.86E-08 | Ru-106 | 2.12E-04 |
| I-130 | 4.41E-02 | Rh-106 | 2.12E-04 |
| I-131 | 3.05E+00 | Ag-110m | 1.99E-03 |
| I-132 | 2.97E+00 | Te-125m | 7.75E-04 |
| I-133 | 4.72E+00 | Te-127m | 3.46E-03 |
| I-134 | 6.49E-01 | Te-127 | 1.43E-02 |
| I-135 | 2.59E+00 | Te-129m | 1.17E-02 |
| Cs-134 | 3.22E+00 | Te-129 | 1.46E-02 |
| Cs-136 | 3.90E+00 | Te-131m | 2.68E-02 |
| Cs-137 | 2.27E+00 | Te-131 | 1.40E-02 |
| Cs-138 | 1.06E+00 | Te-132 | 3.15E-01 |
| Mn-54 | 1.60E-03 | Te-134 | 3.15E-02 |
| H-3 | 3.24E+00 | Ba-137m | 2.15E+00 |
| Cr-51 | 5.40E-03 | Ba-140 | 4.43E-03 |
| Mn-56 | 2.20E-02 | La-140 | 1.52E-03 |
| Fe-55 | 2.10E-03 | Ce-141 | 6.80E-04 |
| Fe-59 | 5.10E-04 | Ce-143 | 5.41E-04 |
| Co-58 | 1.40E-02 | Pr-143 | 6.55E-04 |
| Co-60 | 1.30E-03 | Ce-144 | 5.14E-04 |
| Rb-86 | 3.76E-02 | Pr-144 | 5.14E-04 |

**FIGURE 4-1 RCS LETDOWN PIPING NEAR R-9**

(Reference 6.4)



## 5 ASSUMPTIONS

5.1. It is assumed that the R-9 detector is position 6 inches from the RCS letdown line. This assumption is justified by the original design specifications (References 6.14 and 6.15).

5.2. It is assumed that the DE I-131 concentration for 0.1% failed fuel is 1.6 uCi/gm. This is based on assuming that 0.1% failed fuel, equivalent to about 22 rods (0.001*121*179), releases its iodine gap activity to the RCS and mixes completely. Depending on the gas gap activity, the dose equivalent iodine will vary. However the value is expected to be greater than 1.0 uCi/gm. The value of 1.6 uCi/gm is supported by using Design Inputs 4.3, 4.4 and 4.8 above and is shown in the attached spreadsheet. *It is noted that the assumed value of 1.6 uCi/gm is based on 35,000 MWd/MTU fuel* burnup and core inventory pre-EPU. These values were chosen to represent the conditions most likely associated with a 0.1% gap release and to benchmark the UFSAR R-9 setpoint. With the EPU source term the DE I-131 becomes about 2 uCi/gm.

5.3. Decay due to transit from the reactor to R-9 is not considered. This is reasonable since most isotopes are long-lived. There are only 5 out of 72 isotopes with a half-life of less than 15 minutes. Also, N-16 (half-life of 7.3 seconds) is not modeled since it is expected to decay many half-lives before reaching R-9.

5.4. The density of water is assumed to be 1.0 g/cc. This is done for simplicity and has minimal impact on the results.

5.5. It is assumed that 1 rem equals 1 rad, which is a good assumption for gamma radiation.

5.6. Shielding model features such as pipe insulation and concrete are neglected. This assumption should have a minimal impact on the results. The concrete behind the detector and pipes would provide backscatter resulting in an increase in lower energy photons in the detector. This would result in a slight under-prediction in the monitor response.

5.7. SCALE standard composition library is used for pipe material. The pipes are assumed to be SS304. ANSI/ANS-6.6.1 (Reference 6.16) dry air composition is assumed.

## 6 REFERENCES

6.1 P-9, "Radiation Monitoring System."

6.2 NEI 99-01, Rev. 5, "Methodology for Development of Emergency Action Levels."

6.3 CN-REA-04-34, Rev. 0, "R. E. Ginna RCS Sources for 1811 MWt Uprate."

6.4 C-381-357 (C381-357,2 controlled docno), Rev. 6, "Chemical and Volume Control System from Non-Regenerative HTX to Wall Enclosure (EWR-2512)."

6.5 DA-NS-08-049, Rev. 0, Ginna Gas Gap Isotopic Fraction Calculations."

6.6 ICRP-30, "Limits for Intakes of Radionuclides by Workers," 1979.

6.7 CHI-PRI-EBAR, Rev. 01602, "EBAR, Total Activity and Maximum Activity in the Primary Coolant."

6.8 ICRP-38, "Radionuclide Transformations – Energy and Intensity of Emissions," 1983.

6.9 CN-REA-04-57, Rev. 1, "R. E. Ginna Normal Operation RCS and Secondary Coolant Sources for 1811 MWt Uprate."

6.10 CN-REA-04-32, Rev. 0, "Ginna 1811 MW Uprate – Core, Fuel Handling Accident, and POST/FIPCO ORIGEN Sources."

6.11 CN-REA-02-33, Rev. 0, "Radiation Source Terms for R. E. Ginna Alternate Source Term Analysis."

6.12 "MCNP/MCNPX – Monte Carlo N-Particle Transport Code System Including MCNP5 1.40 and MCNPX 2.4.0," Oak Ridge National Laboratory, RSICC Computer Code Collection CCC-730.

6.13 "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations for Workstations and Personal Computers," Oak Ridge National Laboratory, RSICC Computer Code Collection, NUREG/CR-0200, Revision 6, ORNL/NUREG/CSD-2/V2/R6.

6.14 RGE ECN-70070, "Failed Fuel Element Detector." (see Appendix Section 14).

6.15 Westinghouse Ltr SA-RA-C-43 (Ginna Docno WPLREC-19990719-04721), "Functional Requirements of a Failed Fuel Detector," dated May 26, 1969.

6.16 ANSI/ANS-6.6.1, "Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants."

## 7 DOCUMENTATION OF COMPUTER CODES

ORIGEN-S (Reference 6.13) and MCNP5 (Reference 6.12) were used in this calculation. Spreadsheets were used for input preparation and data reduction.

Computer files associated with this calculation are listed in Table 7-1. The files are stored in electronic format in the configuration management system. No CD-ROMs are part of this calculation.

**TABLE 7-1 LISTING OF COMPUTER FILES**

| | | | |
|---|---|---|---|
| 12/19/2011 | 03:25 AM | 593,815 | CALC-2011-0019 REV000.xlsx |
| 12/19/2011 | 02:31 AM | 5,562 | r9 |
| 12/19/2011 | 02:52 AM | 38,904 | r9o |
| 12/19/2011 | 02:52 AM | 2,658,640 | r9r |
| 12/19/2011 | 02:57 AM | 3,630 | s0p1failed.inp |
| 12/19/2011 | 02:57 AM | 61,472 | s0p1failed.out |
| 12/19/2011 | 02:58 AM | 3,630 | sALERT.inp |
| 12/19/2011 | 02:58 AM | 61,568 | sALERT.out |
| 12/19/2011 | 02:57 AM | 3,630 | sUE.inp |
| 12/19/2011 | 02:57 AM | 61,568 | sUE.out |
| 12/19/2011 | 02:58 AM | 3,630 | sUE_16DEI.inp |
| 12/19/2011 | 02:58 AM | 61,472 | sUE_16DEI.out |

## 8 METHOD OF ANALYSIS

There are two steps in the method of analysis for this calculation. In the first step, the gamma source term is calculated from nuclide specific activities. ORIGEN-S (Reference 6.13) is used to convert nuclide inventories to gamma photon sources. The nuclide inventories are input as concentrations in units of Ci/gm to produce gamma source terms in units of photons/sec/gm. This gamma source concentration is converted to volumetric concentration by multiplying by the density of water.

The second step is the calculation of the dose rates at the R-9 detector. An MCNP (Reference 6.12) model of the letdown piping and the detector arrangement is constructed. A uniform source, by energy group, is applied to calculate the dose rate for each energy group. The ORIGEN-S output, the MCNP output and the source volume are multiplied to obtain the R-9 dose rate.

In order to properly scale the source terms, the methods of calculating the DE I-131 and the 100/EBAR concentrations are required. These methods are disclosed in CHI-PRI-EBAR, but are repeated here. DE I-131 is the equivalent amount of I-131 that corresponds to the activity of a set of iodine isotopes, typically I-131, I-132, I-133, I-134 and I-135, on a dose basis. In other words, it is the dose-weighted-average I-131 of the activity of a set of iodine isotopes. It is calculated by the general expression below:

$$DE\ I-131 = \frac{\sum_i A_i * DCF_i}{DCF_{I-131}} = \sum_i A_i * f_i,\ where\ f_i = \frac{DCF_i}{DCF_{I-131}}.$$

In the above expression, $A_i$ is the activity concentration and $DCF_i$ is the dose conversion factor.

The 100/EBAR applies to the non-iodine isotopes in the reactor coolant. This is calculated in a similar way, but instead the average energy per disintegration (hence the EBAR) is calculated for the RCS mix. The resultant EBAR is then used to calculate the allowable total of the non-iodine isotopes. Per the TS 3.4.16, the total concentration of the non-iodine isotopes is limited to 100/EBAR in uCi/gm. EBAR is calculated as follows:

$$EBAR = \frac{\sum_i A_i * E_i}{A_{total}} = \sum_i A_i * g_i, \ where \ g_i = \frac{A_i}{A_{total}}.$$

In the above expression, $E_i$ is the total (beta + gamma) energy released per disintegration for the non-iodine isotopes.

## 9 ACCEPTANCE CRITERIA

There are no acceptance criteria for this calculation. The results are used to define the technical bases for EALs in the Emergency Plan.

## 10 CALCULATIONS/COMPUTATIONS

### 10.1 SOURCE TERM CALCULATIONS

There are three RCS source terms considered in this calculation:

- 0.1% failed fuel
- 1% failed fuel corresponding to 100/EBAR and DE I-131 of 60 uCi/gm (TS 3.4.16 limiting RCS activity)
- 5% failed fuel corresponding 5x the 100/EBAR concentration and DE I-131 of 300 uCI/gm

The 0.1% failed fuel case is not necessary for EAL calculations; however it was considered for benchmarking purposes. The historical basis for the R-9 high alarm setpoint (Reference 6.1), as indicated in UFSAR 9.3.4.4.9.3, is the dose rate that corresponds to approximately 0.1% fuel rod cladding defects. The high alarm setpoint is 200 mrem/hr. The DE I-131 calculated for the RCS design coolant activity in Table 4-1 is about 4 uCi/gm and therefore one would expect the 0.1% failed fuel DE I-131 to be at least 0.4 uCi/gm. For the 0.1% failed fuel case the DE I-131 modeled is 1.6 uCi/gm and is based on Assumption 5.2.

The 1% and 5% failed fuel cases correspond to the technical bases behind UE and Alert EALs as defined in NEI 99-03 Rev. 5. More specifically, the technical guidance in the determination of these EALs is provided in Appendix Section 13.

EBAR was calculated in the spreadsheet for the RCS design concentrations. The results indicate that the RCS design activity was within about 95% of the 100/EBAR limit, which is expected since the 100/EBAR limit corresponds to 1% failed fuel. Accordingly, the non-iodine isotopes (including activation/corrosion products) were adjusted up by this small difference.

The 1% failed fuel is the base case because it historically is the basis for the maximum design RCS coolant activity in PWRs. From this case, the others are determined. The 5% failed fuel case is simply the 1% failed fuel case activities scaled up by the factor of 5x, with the exception of activation/corrosion products. Since those isotopes are considered independent of the failed fuel, those are left the same.

For the 0.1% case an analogous calculation is performed, with the exception of the iodines as indicated previously. In other words, the non-iodine and non-corrosion/activation products are scaled by the factor of 1/10x.

One additional case was performed for sensitivity purposes. This case is the same as the 1% failed fuel case, except the DE I-131 is adjusted to 16 uCi/gm. This case was made to test the sensitivity of the iodine source term on the R-9 dose rate. The ORIGEN-S runs made are identified in the Table 11-3. Appendix Sections 12.1 and 12.2 contain the ORIGEN-S inputs and selected outputs, respectively.

### 10.2 DOSE RATE CALCULATIONS

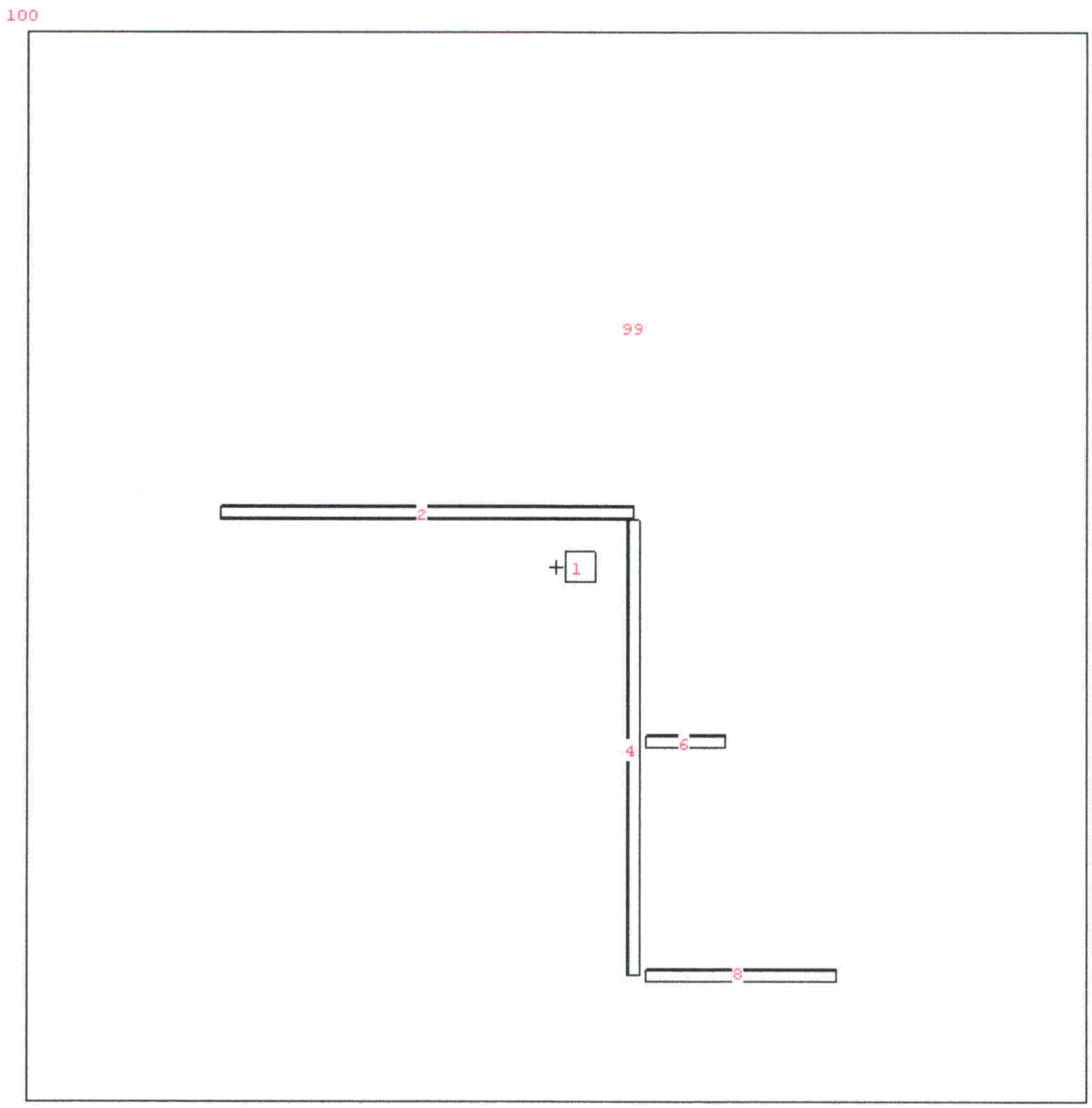
An MCNP model was used to calculate the dose rates at the R-9 detector location. The geometry of the model is shown in Figure 10-1. This model is based on the geometry of Design Input 4.2and is also based on the positioning established in Assumption 5.1. Note, Figure 10-2 is a photograph of the detector for comparison to the MCNP model.

A total of 4 letdown pipe segments were modeled. These pipes represent the line spec 601R piping and the most dose-significant piping in the area. RCC macrobodies were used to model the piping and the ends are unshielded. The unshielded ends are not important since none of them are facing or are in close proximity to the detector.

A tally cell of air is used to represent the R-9 detector volume. A volume flux tally is used on the volume and an energy dependent dose function is used with the flux-to-dose factors identified in Design Input

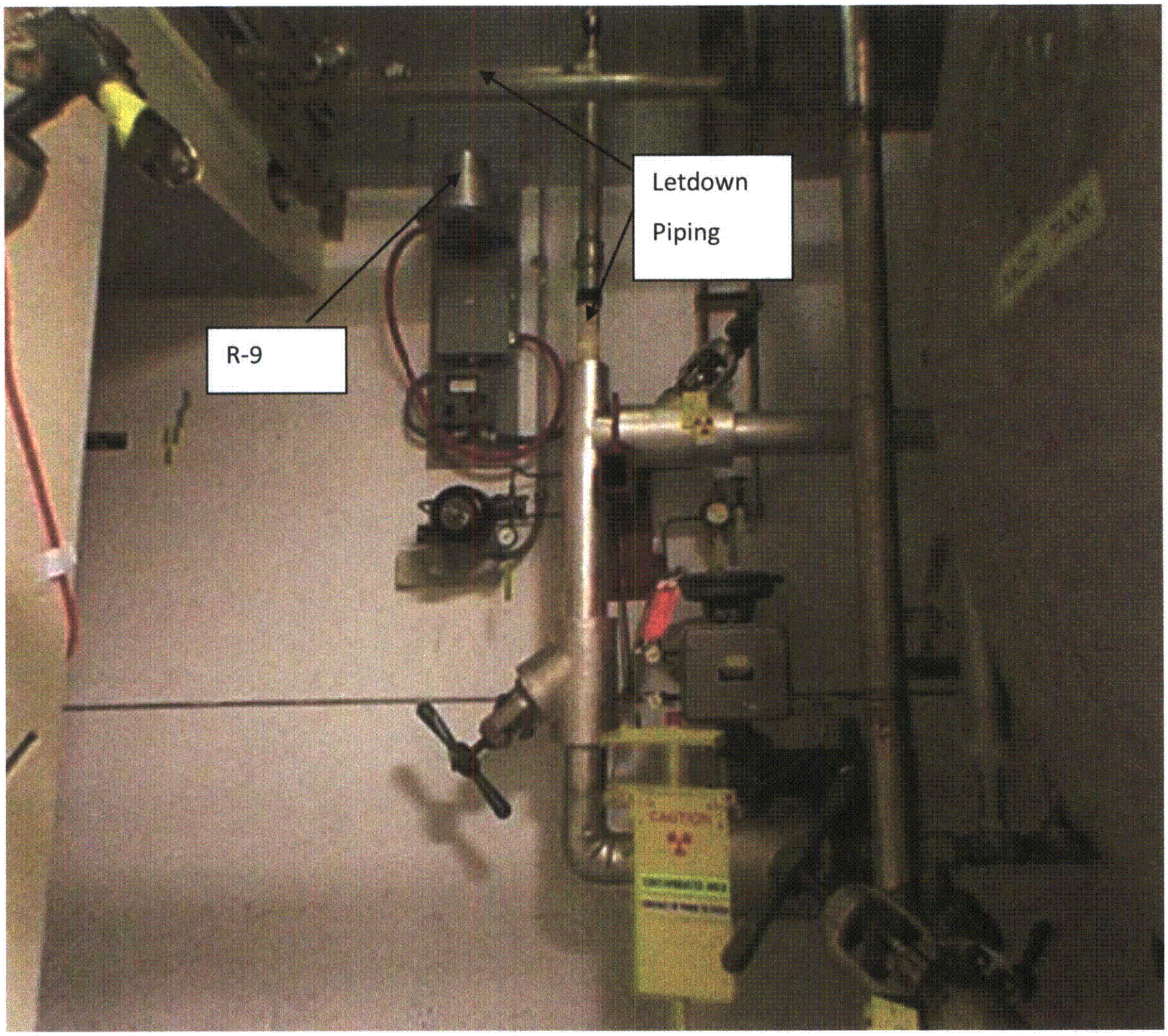
4.9. A standard CASK-81 energy group is used to represent the source and is sampled using a probability of 1.0 for each energy group. The dose tally is treated with the special treatments input to bin the response by energy group (SCX n option). To get the absolute value correct, the tally is multiplied by 18 in the tally multiplier card. Additionally, the tally is multiplied by the volume of the source piping so that resulting dose rates are mrem-cc/hr. This allows easy multiplication of the ORIGEN-S output which is photon/sec-cc to obtain fluxes in the correct units.

**FIGURE 10-1 MCNP GEOMETRY PLOT**




The above plot is a representation of the MCNP model in Section 12.3

**FIGURE 10-2 PHOTOGRAPH OF R-9**



## 11 RESULTS

### 11.1 RCS SOURCE TERMS

The RCS source term inventories are shown in Table 11-1. The column headings in Table 11-1 correspond to the ORIGEN-S run IDs.

**TABLE 11-1 RCS SOURCE TERMS FOR VARIOUS FAILED FUEL SCENARIOS**

| | RCS Activity Concentration (uCi/gm) | | | |
|---|---|---|---|---|
| Nuclide | s0p1failed | sUE | sUE_16DEI | sALERT |
| Kr-83m | 4.99E-02 | 4.99E-01 | 4.99E-01 | 2.49E+00 |
| Kr-85m | 2.03E-01 | 2.03E+00 | 2.03E+00 | 1.02E+01 |
| Kr-85 | 8.64E-01 | 8.64E+00 | 8.64E+00 | 4.32E+01 |
| Kr-87 | 1.30E-01 | 1.30E+00 | 1.30E+00 | 6.52E+00 |
| Kr-88 | 3.79E-01 | 3.79E+00 | 3.79E+00 | 1.89E+01 |
| Kr-89 | 1.05E-02 | 1.05E-01 | 1.05E-01 | 5.26E-01 |
| Xe-131m | 3.72E-01 | 3.72E+00 | 3.72E+00 | 1.86E+01 |
| Xe-133m | 4.04E-01 | 4.04E+00 | 4.04E+00 | 2.02E+01 |
| Xe-133 | 2.85E+01 | 2.85E+02 | 2.85E+02 | 1.43E+03 |
| Xe-135m | 5.87E-02 | 5.87E-01 | 5.87E-01 | 2.93E+00 |
| Xe-135 | 9.98E-01 | 9.98E+00 | 9.98E+00 | 4.99E+01 |
| Xe-137 | 2.01E-02 | 2.01E-01 | 2.01E-01 | 1.00E+00 |
| Xe-138 | 7.28E-02 | 7.28E-01 | 7.28E-01 | 3.64E+00 |
| Br-83 | 1.05E-02 | 1.05E-01 | 1.05E-01 | 5.26E-01 |
| Br-84 | 5.15E-03 | 5.15E-02 | 5.15E-02 | 2.58E-01 |
| Br-85 | 6.00E-04 | 6.00E-03 | 6.00E-03 | 3.00E-02 |
| I-129 | 2.78E-08 | 1.04E-06 | 2.78E-07 | 5.22E-06 |
| I-130 | 1.79E-02 | 6.71E-01 | 1.79E-01 | 3.36E+00 |
| I-131 | 1.24E+00 | 4.64E+01 | 1.24E+01 | 2.32E+02 |
| I-132 | 1.21E+00 | 4.52E+01 | 1.21E+01 | 2.26E+02 |
| I-133 | 1.92E+00 | 7.19E+01 | 1.92E+01 | 3.59E+02 |
| I-134 | 2.63E-01 | 9.88E+00 | 2.63E+00 | 4.94E+01 |
| I-135 | 1.05E+00 | 3.94E+01 | 1.05E+01 | 1.97E+02 |
| Cs-134 | 3.39E-01 | 3.39E+00 | 3.39E+00 | 1.69E+01 |
| Cs-136 | 4.10E-01 | 4.10E+00 | 4.10E+00 | 2.05E+01 |
| Cs-137 | 2.39E-01 | 2.39E+00 | 2.39E+00 | 1.19E+01 |
| Cs-138 | 1.11E-01 | 1.11E+00 | 1.11E+00 | 5.57E+00 |
| Mn-54 | 1.68E-03 | 1.68E-03 | 1.68E-03 | 1.68E-03 |
| H-3 | 3.41E+00 | 3.41E+00 | 3.41E+00 | 3.41E+00 |
| Cr-51 | 5.68E-03 | 5.68E-03 | 5.68E-03 | 5.68E-03 |
| Mn-56 | 2.31E-02 | 2.31E-02 | 2.31E-02 | 2.31E-02 |
| Fe-55 | 2.21E-03 | 2.21E-03 | 2.21E-03 | 2.21E-03 |
| Fe-59 | 5.36E-04 | 5.36E-04 | 5.36E-04 | 5.36E-04 |
| Co-58 | 1.47E-02 | 1.47E-02 | 1.47E-02 | 1.47E-02 |
| Co-60 | 1.37E-03 | 1.37E-03 | 1.37E-03 | 1.37E-03 |
| Rb-86 | 3.96E-03 | 3.96E-02 | 3.96E-02 | 1.98E-01 |

| Nuclide | RCS Activity Concentration (uCi/gm) | | | |
|---|---|---|---|---|
| | s0p1failed | sUE | sUE_16DEI | sALERT |
| Rb-88 | 4.63E-01 | 4.63E+00 | 4.63E+00 | 2.31E+01 |
| Rb-89 | 2.10E-02 | 2.10E-01 | 2.10E-01 | 1.05E+00 |
| Sr-89 | 4.80E-04 | 4.80E-03 | 4.80E-03 | 2.40E-02 |
| Sr-90 | 2.45E-05 | 2.45E-04 | 2.45E-04 | 1.23E-03 |
| Sr-91 | 6.31E-04 | 6.31E-03 | 6.31E-03 | 3.16E-02 |
| Sr-92 | 1.39E-04 | 1.39E-03 | 1.39E-03 | 6.94E-03 |
| Y-90 | 7.03E-06 | 7.03E-05 | 7.03E-05 | 3.51E-04 |
| Y-91m | 3.43E-04 | 3.43E-03 | 3.43E-03 | 1.71E-02 |
| Y-91 | 6.31E-05 | 6.31E-04 | 6.31E-04 | 3.16E-03 |
| Y-92 | 1.22E-04 | 1.22E-03 | 1.22E-03 | 6.10E-03 |
| Y-93 | 4.17E-05 | 4.17E-04 | 4.17E-04 | 2.08E-03 |
| Zr-95 | 7.35E-05 | 7.35E-04 | 7.35E-04 | 3.68E-03 |
| Nb-95 | 7.40E-05 | 7.40E-04 | 7.40E-04 | 3.70E-03 |
| Mo-99 | 8.82E-02 | 8.82E-01 | 8.82E-01 | 4.41E+00 |
| Tc-99m | 8.18E-02 | 8.18E-01 | 8.18E-01 | 4.09E+00 |
| Ru-103 | 6.43E-05 | 6.43E-04 | 6.43E-04 | 3.21E-03 |
| Rh-103m | 6.46E-05 | 6.46E-04 | 6.46E-04 | 3.23E-03 |
| Ru-106 | 2.23E-05 | 2.23E-04 | 2.23E-04 | 1.12E-03 |
| Rh-106 | 2.23E-05 | 2.23E-04 | 2.23E-04 | 1.12E-03 |
| Ag-110m | 2.09E-04 | 2.09E-03 | 2.09E-03 | 1.05E-02 |
| Te-125m | 8.15E-05 | 8.15E-04 | 8.15E-04 | 4.08E-03 |
| Te-127m | 3.64E-04 | 3.64E-03 | 3.64E-03 | 1.82E-02 |
| Te-127 | 1.50E-03 | 1.50E-02 | 1.50E-02 | 7.52E-02 |
| Te-129m | 1.23E-03 | 1.23E-02 | 1.23E-02 | 6.15E-02 |
| Te-129 | 1.54E-03 | 1.54E-02 | 1.54E-02 | 7.68E-02 |
| Te-131m | 2.82E-03 | 2.82E-02 | 2.82E-02 | 1.41E-01 |
| Te-131 | 1.47E-03 | 1.47E-02 | 1.47E-02 | 7.36E-02 |
| Te-132 | 3.31E-02 | 3.31E-01 | 3.31E-01 | 1.66E+00 |
| Te-134 | 3.31E-03 | 3.31E-02 | 3.31E-02 | 1.66E-01 |
| Ba-137m | 2.26E-01 | 2.26E+00 | 2.26E+00 | 1.13E+01 |
| Ba-140 | 4.66E-04 | 4.66E-03 | 4.66E-03 | 2.33E-02 |
| La-140 | 1.60E-04 | 1.60E-03 | 1.60E-03 | 7.99E-03 |
| Ce-141 | 7.15E-05 | 7.15E-04 | 7.15E-04 | 3.58E-03 |
| Ce-143 | 5.69E-05 | 5.69E-04 | 5.69E-04 | 2.85E-03 |
| Pr-143 | 6.89E-05 | 6.89E-04 | 6.89E-04 | 3.45E-03 |
| Ce-144 | 5.41E-05 | 5.41E-04 | 5.41E-04 | 2.70E-03 |
| Pr-144 | 5.41E-05 | 5.41E-04 | 5.41E-04 | 2.70E-03 |

## 11.2 R-9 RELATIVE DOSE RATES

The relative dose rates for the R-9 detector are summarized in Table 11-2. To utilize these results convert an activity concentration to photon/sec per cc according to the energy group structure specified and multiply the values in Table 11-2 to get the dose rate in mrem/hr.

**TABLE 11-2 R-9 RELATIVE DOSE RATES**

| | ENERGY (MEV) | | Dose Rate | Relative |
|---|---|---|---|---|
| FROM | TO | AVE | (mrem-cc/hr) | Error |
| 1.00E-02 | 5.00E-02 | 3.00E-02 | 1.39E-11 | 0.2357 |
| 5.00E-02 | 1.00E-01 | 7.50E-02 | 7.30E-09 | 0.0494 |
| 1.00E-01 | 2.00E-01 | 1.50E-01 | 5.32E-08 | 0.0216 |
| 2.00E-01 | 3.00E-01 | 2.50E-01 | 1.17E-07 | 0.018 |
| 3.00E-01 | 4.00E-01 | 3.50E-01 | 1.76E-07 | 0.0171 |
| 4.00E-01 | 6.00E-01 | 5.00E-01 | 2.47E-07 | 0.0168 |
| 6.00E-01 | 8.00E-01 | 7.00E-01 | 3.20E-07 | 0.0167 |
| 8.00E-01 | 1.00E+00 | 9.00E-01 | 4.04E-07 | 0.0165 |
| 1.00E+00 | 1.33E+00 | 1.17E+00 | 4.99E-07 | 0.0163 |
| 1.33E+00 | 1.66E+00 | 1.50E+00 | 5.79E-07 | 0.0165 |
| 1.66E+00 | 2.00E+00 | 1.83E+00 | 7.12E-07 | 0.0161 |
| 2.00E+00 | 2.50E+00 | 2.25E+00 | 8.17E-07 | 0.0162 |
| 2.50E+00 | 3.00E+00 | 2.75E+00 | 9.46E-07 | 0.016 |
| 3.00E+00 | 4.00E+00 | 3.50E+00 | 1.08E-06 | 0.0163 |
| 4.00E+00 | 5.00E+00 | 4.50E+00 | 1.29E-06 | 0.0161 |
| 5.00E+00 | 6.50E+00 | 5.75E+00 | 1.58E-06 | 0.0158 |
| 6.50E+00 | 8.00E+00 | 7.25E+00 | 1.85E-06 | 0.0158 |
| 8.00E+00 | 1.00E+01 | 9.00E+00 | 2.21E-06 | 0.0157 |

### 11.3 R-9 DOSE RATES DUE TO FUEL FAILURE

The calculated R-9 radiation monitor dose rates for the scenarios analyzed are summarized in Table 11-3.

The 0.1% result shows reasonable agreement to the historical alarm setpoint basis of 200 mrem/hr. This result is predicated on the assumption that the iodine activity is 1.6 uCi/gm DE I-131. Although Assumption 5.2 establishes a basis for this, a change in the DE I-131 will change the results considerably. For example, with the EPU core source term the 1.6 uCi/gm increases to approximately 2 uCi/gm, which would most likely lead to an R-9 response greater than 200 mrem/hr.

The 1% and 60 uCi/gm DE I-131 case establishes the basis for the UE EAL. This case shows the significance of the iodine to the R-9 response as the dose rate increases almost by 30x even though the non-iodine fission products only increases 10x. A tenfold (9.8) increase of the 0.1% case is shown at the bottom of Table 11-3 when the iodine activity concentration increases by the same amount. The DE I-131 of 60 uCi/gm is used because the NEI 99-01 Rev. 5 guidance specifies the UE occurs when the coolant activity exceeds "technical specifications for transient iodine spiking limits".

The last case shows the results when the DE I-131 is 300 uCi/gm. According to NEI 99-01 Rev. 5, "this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage." The NEI 99-01 Rev. 5 goes on to say that this RCS activity level indicates "significant clad damage and thus the Fuel Clad Barrier is considered lost."

The results for the 300 uCl/gm DE I-131 case are rather high and exceed the upper range identified in the UFSAR 9.3.4.4.9.3. Nevertheless, they are still included in case the use of R-9 or other detector is desired for the Alert EAL.

Note the RCS source terms are included in Appendix Section 12.2.

**TABLE 11-3 R-9 DOSE RATES DUE TO FUEL FAILURE**

| Fuel Failure | DE I-131 (uCi/gm) | Frac 100/EBAR | R-9 Dose Rate (mrem/hr) | ORIGEN-S Case |
|---|---|---|---|---|
| 0.10% | 1.6 | 0.1 | 177.26 | s0p1failed |
| 1% | 60 | 1 | 4855.15 | sUE |
| 5% | 300 | 5 | 24253.97 | sALERT |
| 1% | 16 | 1 | 1739.46 | sUE_16DEI |

# 12 APPENDIX: COMPUTER FILES

## 12.1 ORIGEN-S INPUT FILES

**s0p1failed**

```
#ORIGENS
0$$   E T
DECAY CASE
3$$ 21 1 1 0 A16 4  A33 18  E T
35$$ 0 T
54$$ A8 1  E
56$$ A2 1 A6 1  A10  0  A13 74  A14 3  A15 3 E
57** 0 E T
CONVERSION OF ACTIVITIES TO ENERGY GROUPS
BASIS
60**
   0.0000E+00 E
61** F1e-20
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
 1  0  0      1  0  0    1  0  0    1  0  0       3Z         6Z
 1  0  0      1  0  0    1  0  0    1  0  0       3Z         6Z
 1  0  0      1  0  0    1  0  0    1  0  0       3Z         6Z
81$$ 2 0 26 1  E
82$$ F2
83**   1.0000E+07  8.0000E+06  6.5000E+06  5.0000E+06  4.0000E+06
   3.0000E+06  2.5000E+06  2.0000E+06  1.6600E+06  1.3300E+06
   1.0000E+06  8.0000E+05  6.0000E+05  4.0000E+05  3.0000E+05
   2.0000E+05  1.0000E+05  5.0000E+04  1.0000E+04
73$$  110240 360831 360851 360850 360870
  360880 360890 541311 541331 541330
  541351 541350 541370 541380 350830
  350840 350850 531290 531300 531310
  531320 531330 531340 531350 551340
  551360 551370 551380 250540 010030
  240510 250560 260550 260590 270580
  270600 370860 370880 370890 380890
  380900 380910 380920 390900 390911
  390910 390920 390930 400950 410950
  420990 430991 441030 451031 441060
  451060 471101 521251 521271 521270
  521291 521290 521311 521310 521320
  521340 561371 561400 571400 581410
  581430 591430 581440 591440
74**   5.0000E-08  4.9900E-08  2.0300E-07  8.6400E-07  1.3000E-07
   3.7900E-07  1.0500E-08  3.7200E-07  4.0400E-07  2.8500E-05
   5.8700E-08  9.9800E-07  2.0100E-08  7.2800E-08  1.0500E-08
   5.1500E-09  6.0000E-10  2.7800E-14  1.7900E-08  1.2400E-06
   1.2100E-06  1.9200E-06  2.6300E-07  1.0500E-06  3.3900E-07
   4.1000E-07  2.3900E-07  1.1100E-07  1.6800E-09  3.4100E-06
   5.6800E-09  2.3100E-08  2.2100E-09  5.3600E-10  1.4700E-08
   1.3700E-09  3.9600E-09  4.6300E-07  2.1000E-08  4.8000E-10
   2.4500E-11  6.3100E-10  1.3900E-10  7.0300E-12  3.4300E-10
   6.3100E-11  1.2200E-10  4.1700E-11  7.3500E-11  7.4000E-11
   8.8200E-08  8.1800E-08  6.4300E-11  6.4600E-11  2.2300E-11
   2.2300E-11  2.0900E-10  8.1500E-11  3.6400E-10  1.5000E-09
   1.2300E-09  1.5400E-09  2.8200E-09  1.4700E-09  3.3100E-08
   3.3100E-09  2.2600E-07  4.6600E-10  1.6000E-10  7.1500E-11
   5.6900E-11  6.8900E-11  5.4100E-11  5.4100E-11
```

```
75$$  1 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 1 3
  1 1 1 1 1 1 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3
       3 T
@DECAY STEP  1 :    0.0000E+00
56$$ F0 T
END
```

**sUE**

```
#ORIGENS
0$$   E T
DECAY CASE
3$$ 21 1 1 0 A16 4  A33 18  E T
35$$ 0 T
54$$ A8 1  E
56$$ A2 1 A6 1  A10  0  A13 74  A14 3  A15 3 E
57** 0 E T
CONVERSION OF ACTIVITIES TO ENERGY GROUPS
BASIS
60**
   0.0000E+00 E
61** F1e-20
65$$
'GRAM-ATOMS   GRAMS    CURIES    WATTS-ALL  WATTS-GAMMA
 1  0  0     1  0  0    1  0  0    1  0  0       3Z          6Z
 1  0  0     1  0  0    1  0  0    1  0  0       3Z          6Z
 1  0  0     1  0  0    1  0  0    1  0  0       3Z          6Z
81$$ 2 0 26 1  E
82$$ F2
83**    1.0000E+07  8.0000E+06  6.5000E+06  5.0000E+06  4.0000E+06
   3.0000E+06  2.5000E+06  2.0000E+06  1.6600E+06  1.3300E+06
   1.0000E+06  8.0000E+05  6.0000E+05  4.0000E+05  3.0000E+05
   2.0000E+05  1.0000E+05  5.0000E+04  1.0000E+04
73$$  110240 360831 360851 360850 360870
  360880 360890 541311 541331 541330
  541351 541350 541370 541380 350830
  350840 350850 531290 531300 531310
  531320 531330 531340 531350 551340
  551360 551370 551380 250540 010030
  240510 250560 260550 260590 270580
  270600 370860 370880 370890 380890
  380900 380910 380920 390900 390911
  390910 390920 390930 400950 410950
  420990 430991 441030 451031 441060
  451060 471101 521251 521271 521270
  521291 521290 521311 521310 521320
  521340 561371 561400 571400 581410
  581430 591430 581440 591440
74**    5.0000E-08  4.9900E-07  2.0300E-06  8.6400E-06  1.3000E-06
   3.7900E-06  1.0500E-07  3.7200E-06  4.0400E-06  2.8500E-04
   5.8700E-07  9.9800E-06  2.0100E-07  7.2800E-07  1.0500E-07
   5.1500E-08  6.0000E-09  1.0400E-12  6.7100E-07  4.6400E-05
   4.5200E-05  7.1900E-05  9.8800E-06  3.9400E-05  3.3900E-06
   4.1000E-06  2.3900E-06  1.1100E-06  1.6800E-09  3.4100E-06
   5.6800E-09  2.3100E-08  2.2100E-09  5.3600E-10  1.4700E-08
   1.3700E-09  3.9600E-08  4.6300E-06  2.1000E-07  4.8000E-09
```

```
   2.4500E-10  6.3100E-09  1.3900E-09  7.0300E-11  3.4300E-09
   6.3100E-10  1.2200E-09  4.1700E-10  7.3500E-10  7.4000E-10
   8.8200E-07  8.1800E-07  6.4300E-10  6.4600E-10  2.2300E-10
   2.2300E-10  2.0900E-09  8.1500E-10  3.6400E-09  1.5000E-08
   1.2300E-08  1.5400E-08  2.8200E-08  1.4700E-08  3.3100E-07
   3.3100E-08  2.2600E-06  4.6600E-09  1.6000E-09  7.1500E-10
   5.6900E-10  6.8900E-10  5.4100E-10  5.4100E-10
75$$  1 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 1 3
  1 1 1 1 1 1 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3
       3 T
@DECAY STEP  1 :   0.0000E+00
56$$ F0 T
END
```

**sALERT**

```
#ORIGENS
0$$   E T
DECAY CASE
3$$ 21 1 1 0 A16 4  A33 18  E T
35$$ 0 T
54$$ A8 1  E
56$$ A2 1 A6 1  A10  0  A13 74  A14 3  A15 3 E
57** 0 E T
CONVERSION OF ACTIVITIES TO ENERGY GROUPS
BASIS
60**
   0.0000E+00 E
61** F1e-20
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
 1  0  0     1  0  0    1  0  0    1  0  0       3Z         6Z
 1  0  0     1  0  0    1  0  0    1  0  0       3Z         6Z
 1  0  0     1  0  0    1  0  0    1  0  0       3Z         6Z
81$$ 2 0 26 1  E
82$$ F2
83**    1.0000E+07  8.0000E+06  6.5000E+06  5.0000E+06  4.0000E+06
   3.0000E+06  2.5000E+06  2.0000E+06  1.6600E+06  1.3300E+06
   1.0000E+06  8.0000E+05  6.0000E+05  4.0000E+05  3.0000E+05
   2.0000E+05  1.0000E+05  5.0000E+04  1.0000E+04
73$$  110240 360831 360851 360850 360870
  360880 360890 541311 541331 541330
  541351 541350 541370 541380 350830
  350840 350850 531290 531300 531310
  531320 531330 531340 531350 551340
  551360 551370 551380 250540 010030
  240510 250560 260550 260590 270580
  270600 370860 370880 370890 380890
  380900 380910 380920 390900 390911
  390910 390920 390930 400950 410950
  420990 430991 441030 451031 441060
  451060 471101 521251 521271 521270
  521291 521290 521311 521310 521320
  521340 561371 561400 571400 581410
  581430 591430 581440 591440
```

```
74**    5.0000E-08  2.4900E-06  1.0200E-05  4.3200E-05  6.5200E-06
   1.8900E-05  5.2600E-07  1.8600E-05  2.0200E-05  1.4300E-03
   2.9300E-06  4.9900E-05  1.0000E-06  3.6400E-06  5.2600E-07
   2.5800E-07  3.0000E-08  5.2200E-12  3.3600E-06  2.3200E-04
   2.2600E-04  3.5900E-04  4.9400E-05  1.9700E-04  1.6900E-05
   2.0500E-05  1.1900E-05  5.5700E-06  1.6800E-09  3.4100E-06
   5.6800E-09  2.3100E-08  2.2100E-09  5.3600E-10  1.4700E-08
   1.3700E-09  1.9800E-07  2.3100E-05  1.0500E-06  2.4000E-08
   1.2300E-09  3.1600E-08  6.9400E-09  3.5100E-10  1.7100E-08
   3.1600E-09  6.1000E-09  2.0800E-09  3.6800E-09  3.7000E-09
   4.4100E-06  4.0900E-06  3.2100E-09  3.2300E-09  1.1200E-09
   1.1200E-09  1.0500E-08  4.0800E-09  1.8200E-08  7.5200E-08
   6.1500E-08  7.6800E-08  1.4100E-07  7.3600E-08  1.6600E-06
   1.6600E-07  1.1300E-05  2.3300E-08  7.9900E-09  3.5800E-09
   2.8500E-09  3.4500E-09  2.7000E-09  2.7000E-09
75$$  1 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 1 3
  1 1 1 1 1 1 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3
       3 T
@DECAY STEP  1 :    0.0000E+00
56$$ F0 T
END
```

## sALERT DEI16

```
#ORIGENS
0$$   E T
DECAY CASE
3$$ 21 1 1 0 A16 4  A33 18  E T
35$$ 0 T
54$$ A8 1  E
56$$ A2 1 A6 1  A10  0  A13 74  A14 3  A15 3 E
57** 0 E T
CONVERSION OF ACTIVITIES TO ENERGY GROUPS
BASIS
60**
   0.0000E+00 E
61** F1e-20
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
 1  0  0      1  0  0    1  0  0    1  0  0       3Z         6Z
 1  0  0      1  0  0    1  0  0    1  0  0       3Z         6Z
 1  0  0      1  0  0    1  0  0    1  0  0       3Z         6Z
81$$ 2 0 26 1  E
82$$ F2
83**    1.0000E+07  8.0000E+06  6.5000E+06  5.0000E+06  4.0000E+06
   3.0000E+06  2.5000E+06  2.0000E+06  1.6600E+06  1.3300E+06
   1.0000E+06  8.0000E+05  6.0000E+05  4.0000E+05  3.0000E+05
   2.0000E+05  1.0000E+05  5.0000E+04  1.0000E+04
73$$  110240 360831 360851 360850 360870
  360880 360890 541311 541331 541330
  541351 541350 541370 541380 350830
  350840 350850 531290 531300 531310
  531320 531330 531340 531350 551340
  551360 551370 551380 250540 010030
  240510 250560 260550 260590 270580
```

```
  270600 370860 370880 370890 380890
  380900 380910 380920 390900 390911
  390910 390920 390930 400950 410950
  420990 430991 441030 451031 441060
  451060 471101 521251 521271 521270
  521291 521290 521311 521310 521320
  521340 561371 561400 571400 581410
  581430 591430 581440 591440
74**    5.0000E-08  4.9900E-07  2.0300E-06  8.6400E-06  1.3000E-06
   3.7900E-06  1.0500E-07  3.7200E-06  4.0400E-06  2.8500E-04
   5.8700E-07  9.9800E-06  2.0100E-07  7.2800E-07  1.0500E-07
   5.1500E-08  6.0000E-09  2.7800E-13  1.7900E-07  1.2400E-05
   1.2100E-05  1.9200E-05  2.6300E-06  1.0500E-05  3.3900E-06
   4.1000E-06  2.3900E-06  1.1100E-06  1.6800E-09  3.4100E-06
   5.6800E-09  2.3100E-08  2.2100E-09  5.3600E-10  1.4700E-08
   1.3700E-09  3.9600E-08  4.6300E-06  2.1000E-07  4.8000E-09
   2.4500E-10  6.3100E-09  1.3900E-09  7.0300E-11  3.4300E-09
   6.3100E-10  1.2200E-09  4.1700E-10  7.3500E-10  7.4000E-10
   8.8200E-07  8.1800E-07  6.4300E-10  6.4600E-10  2.2300E-10
   2.2300E-10  2.0900E-09  8.1500E-10  3.6400E-09  1.5000E-08
   1.2300E-08  1.5400E-08  2.8200E-08  1.4700E-08  3.3100E-07
   3.3100E-08  2.2600E-06  4.6600E-09  1.6000E-09  7.1500E-10
   5.6900E-10  6.8900E-10  5.4100E-10  5.4100E-10
75$$  1 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 1 3
  1 1 1 1 1 1 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3
  3 3 3
       3 T
@DECAY STEP  1 :    0.0000E+00
56$$ F0 T
END
```

## 12.2 ORIGEN-S OUTPUT SUMMARY

**<u>s0p1failed</u>**

```
the sources include photons of nuclides for...

  light elements
  actinides
  fission products
                  gamma source spectrum for @DECAY STEP  1 :    0.0000E+00

                                       0.00 hour time of the requested nuclides

                      energy interval in mev       photons / second     mev / second
```

| energy interval in mev | | | photons / second | mev / second |
|---|---|---|---|---|
| 1.0000E-02 | to | 5.0000E-02 | 6.3901E+05 | 1.9170E+04 |
| 5.0000E-02 | to | 1.0000E-01 | 4.5617E+05 | 3.4213E+04 |
| 1.0000E-01 | to | 2.0000E-01 | 3.6783E+04 | 5.5174E+03 |
| 2.0000E-01 | to | 3.0000E-01 | 5.0969E+04 | 1.2742E+04 |
| 3.0000E-01 | to | 4.0000E-01 | 5.4061E+04 | 1.8921E+04 |
| 4.0000E-01 | to | 6.0000E-01 | 1.0663E+05 | 5.3315E+04 |
| 6.0000E-01 | to | 8.0000E-01 | 1.2921E+05 | 9.0447E+04 |
| 8.0000E-01 | to | 1.0000E+00 | 6.5094E+04 | 5.8585E+04 |
| 1.0000E+00 | to | 1.3300E+00 | 5.5107E+04 | 6.4199E+04 |

| | | | |
|---|---|---|---|
| 1.3300E+00 to | 1.6600E+00 | 1.8524E+04 | 2.7694E+04 |
| 1.6600E+00 to | 2.0000E+00 | 1.5295E+04 | 2.7990E+04 |
| 2.0000E+00 to | 2.5000E+00 | 1.1935E+04 | 2.6853E+04 |
| 2.5000E+00 to | 3.0000E+00 | 3.4912E+03 | 9.6009E+03 |
| 3.0000E+00 to | 4.0000E+00 | 2.1907E+02 | 7.6673E+02 |
| 4.0000E+00 to | 5.0000E+00 | 3.6685E+01 | 1.6508E+02 |
| 5.0000E+00 to | 6.5000E+00 | 1.6096E-01 | 9.2549E-01 |
| 6.5000E+00 to | 8.0000E+00 | 0.0000E+00 | 0.0000E+00 |
| 8.0000E+00 to | 1.0000E+01 | 0.0000E+00 | 0.0000E+00 |
| totals | | 1.6425E+06 | 4.5018E+05 |

total energy from nuclides with spectrum data = 4.5018E+05

total energy from nuclides with no spectrum data = 1.3626E-01

**sUE**

the sources include photons of nuclides for...

light elements
actinides
fission products

gamma source spectrum for @DECAY STEP 1 : 0.0000E+00

0.00 hour time of the requested nuclides

| energy interval in mev | | photons / second | mev / second |
|---|---|---|---|
| 1.0000E-02 to | 5.0000E-02 | 7.2521E+06 | 2.1756E+05 |
| 5.0000E-02 to | 1.0000E-01 | 4.8510E+06 | 3.6382E+05 |
| 1.0000E-01 to | 2.0000E-01 | 5.5102E+05 | 8.2654E+04 |
| 2.0000E-01 to | 3.0000E-01 | 7.5417E+05 | 1.8854E+05 |
| 3.0000E-01 to | 4.0000E-01 | 1.6782E+06 | 5.8736E+05 |
| 4.0000E-01 to | 6.0000E-01 | 3.4661E+06 | 1.7330E+06 |
| 6.0000E-01 to | 8.0000E-01 | 4.2192E+06 | 2.9534E+06 |
| 8.0000E-01 to | 1.0000E+00 | 1.6840E+06 | 1.5156E+06 |
| 1.0000E+00 to | 1.3300E+00 | 1.5858E+06 | 1.8474E+06 |
| 1.3300E+00 to | 1.6600E+00 | 4.5838E+05 | 6.8527E+05 |
| 1.6600E+00 to | 2.0000E+00 | 4.3064E+05 | 7.8808E+05 |
| 2.0000E+00 to | 2.5000E+00 | 1.6445E+05 | 3.7001E+05 |
| 2.5000E+00 to | 3.0000E+00 | 1.9089E+04 | 5.2495E+04 |
| 3.0000E+00 to | 4.0000E+00 | 2.1668E+03 | 7.5838E+03 |
| 4.0000E+00 to | 5.0000E+00 | 3.6672E+02 | 1.6502E+03 |
| 5.0000E+00 to | 6.5000E+00 | 1.6096E+00 | 9.2549E+00 |
| 6.5000E+00 to | 8.0000E+00 | 0.0000E+00 | 0.0000E+00 |
| 8.0000E+00 to | 1.0000E+01 | 0.0000E+00 | 0.0000E+00 |
| totals | | 2.7117E+07 | 1.1395E+07 |

total energy from nuclides with spectrum data = 1.1395E+07

total energy from nuclides with no spectrum data = 1.3626E-01

**sALERT**

the sources include photons of nuclides for...

light elements
actinides
fission products

gamma source spectrum for @DECAY STEP 1 : 0.0000E+00

0.00 hour time of the requested nuclides

| energy interval in mev | | | photons / second | mev / second |
|---|---|---|---|---|
| 1.0000E-02 | to | 5.0000E-02 | 3.6351E+07 | 1.0905E+06 |
| 5.0000E-02 | to | 1.0000E-01 | 2.4328E+07 | 1.8246E+06 |
| 1.0000E-01 | to | 2.0000E-01 | 2.7547E+06 | 4.1321E+05 |
| 2.0000E-01 | to | 3.0000E-01 | 3.7704E+06 | 9.4261E+05 |
| 3.0000E-01 | to | 4.0000E-01 | 8.3908E+06 | 2.9368E+06 |
| 4.0000E-01 | to | 6.0000E-01 | 1.7311E+07 | 8.6556E+06 |
| 6.0000E-01 | to | 8.0000E-01 | 2.1094E+07 | 1.4766E+07 |
| 8.0000E-01 | to | 1.0000E+00 | 8.4126E+06 | 7.5713E+06 |
| 1.0000E+00 | to | 1.3300E+00 | 7.9275E+06 | 9.2356E+06 |
| 1.3300E+00 | to | 1.6600E+00 | 2.2852E+06 | 3.4164E+06 |
| 1.6600E+00 | to | 2.0000E+00 | 2.1519E+06 | 3.9380E+06 |
| 2.0000E+00 | to | 2.5000E+00 | 8.2077E+05 | 1.8467E+06 |
| 2.5000E+00 | to | 3.0000E+00 | 8.8077E+04 | 2.4221E+05 |
| 3.0000E+00 | to | 4.0000E+00 | 1.0827E+04 | 3.7894E+04 |
| 4.0000E+00 | to | 5.0000E+00 | 1.8307E+03 | 8.2383E+03 |
| 5.0000E+00 | to | 6.5000E+00 | 8.0304E+00 | 4.6175E+01 |
| 6.5000E+00 | to | 8.0000E+00 | 0.0000E+00 | 0.0000E+00 |
| 8.0000E+00 | to | 1.0000E+01 | 0.0000E+00 | 0.0000E+00 |
| totals | | | 1.3570E+08 | 5.6925E+07 |

total energy from nuclides with spectrum data = 5.6925E+07

total energy from nuclides with no spectrum data = 1.3626E-01

**sUE_16DEI**

the sources include photons of nuclides for...

light elements
actinides
fission products

gamma source spectrum for @DECAY STEP. 1 : 0.0000E+00

0.00 hour time of the requested nuclides

| energy interval in mev | | | photons / second | mev / second |
|---|---|---|---|---|
| 1.0000E-02 | to | 5.0000E-02 | 6.3840E+06 | 1.9152E+05 |
| 5.0000E-02 | to | 1.0000E-01 | 4.5596E+06 | 3.4197E+05 |
| 1.0000E-01 | to | 2.0000E-01 | 3.6638E+05 | 5.4957E+04 |
| 2.0000E-01 | to | 3.0000E-01 | 5.0923E+05 | 1.2731E+05 |
| 3.0000E-01 | to | 4.0000E-01 | 5.4011E+05 | 1.8904E+05 |
| 4.0000E-01 | to | 6.0000E-01 | 1.0646E+06 | 5.3230E+05 |
| 6.0000E-01 | to | 8.0000E-01 | 1.2920E+06 | 9.0441E+05 |

| | | | | |
|---|---|---|---|---|
| 8.0000E-01 | to | 1.0000E+00 | 6.3882E+05 | 5.7494E+05 |
| 1.0000E+00 | to | 1.3300E+00 | 5.5032E+05 | 6.4112E+05 |
| 1.3300E+00 | to | 1.6600E+00 | 1.6980E+05 | 2.5384E+05 |
| 1.6600E+00 | to | 2.0000E+00 | 1.5086E+05 | 2.7606E+05 |
| 2.0000E+00 | to | 2.5000E+00 | 1.1831E+05 | 2.6620E+05 |
| 2.5000E+00 | to | 3.0000E+00 | 1.8120E+04 | 4.9830E+04 |
| 3.0000E+00 | to | 4.0000E+00 | 2.1668E+03 | 7.5838E+03 |
| 4.0000E+00 | to | 5.0000E+00 | 3.6672E+02 | 1.6502E+03 |
| 5.0000E+00 | to | 6.5000E+00 | 1.6096E+00 | 9.2549E+00 |
| 6.5000E+00 | to | 8.0000E+00 | 0.0000E+00 | 0.0000E+00 |
| 8.0000E+00 | to | 1.0000E+01 | 0.0000E+00 | 0.0000E+00 |
| totals | | | 1.6365E+07 | 4.4127E+06 |

total energy from nuclides with spectrum data = 4.4127E+06

total energy from nuclides with no spectrum data = 1.3626E-01

## 12.3 MCNP MODEL INPUT FILE

```
GINNA FAILED FUEL RAD MONITIOR R-9
c
c This model calculates the gamma response factors for the R-9 radiation
c monitor at Ginna station.  The radiation monitor is on the let down
c line in the NOAH tank room in the AUXB basement.  The monitor is
c referred to as the "failed fuel" monitor and has a range of 0.1 to 10E7 mR.
c The detector was a G-M tube and is situated about 6" from
c the 2" letdown line.  Drawing C-381-357 (C381-357,2 DOCNO) shows the
c piping at B-8.  R-9 is just upstream of PCV-135.  The piping is line
c spec 601R, has an OD of 2.375" and a wall thickness of .145".
c An approximate size of the detector (outside of cylinder housing) is
c 5.5" diameter and 5.5" high.  The PO for the replacement detector
c is NQ-10037-B-TS.  The detector is assumed to be 10" from
c vertical letdown pipe.
c
c cell cards
c
1   4 -1.225E-3 -1           imp:p=1 $ detector volume
2   1 -1.0         -2        imp:p=1 $ upper pipe
3   2 -7.94     2 -3         imp:p=1 $ upper pipe wall
4   1 -1.0         -4        imp:p=1 $ right pipe
5   2 -7.94     4 -5         imp:p=1 $ right pipe wall
6   1 -1.0         -6        imp:p=1 $ tee pipe
7   2 -7.94     6 -7         imp:p=1 $ tee pipe wall
8   1 -1.0         -8        imp:p=1 $ tee pipe
9   2 -7.94     8 -9         imp:p=1 $ tee pipe wall
99  3 -1.225E-3 -99 +1 +3 +5 +7 +9 imp:p=1 $ air surround
100 0           +99          imp:p=0 $ outside world

c
c surface cards
c
1  rcc  5.08 0 -6.985 0 0 13.97 6.985                    $ detector
2  rcc  -158.93288 0 25.24125 195.25488 0 0 2.64795 $ upper 2" pipe ID
3  rcc  -158.93288 0 25.24125 195.25488 0 0 3.01625 $ upper 2" pipe OD
4  rcc  36.322 0 22.224 0 0 -213.36 2.64795          $ right 2" pipe ID
5  rcc  36.322 0 22.224 0 0 -213.36 3.01625          $ right 2" pipe OD
6  rcc  42.3545 0 -81.43875 37.368 0 0 2.64795       $ tee off upright piece ID
7  rcc  42.3545 0 -81.43875 37.368 0 0 3.01625       $ tee off upright piece OD
8  rcc  42.3545 0 -191.136 89.52 0 0 2.64795         $ bottom horizontal ID
```

```
9  rcc  42.3545 0 -191.136 89.52 0 0 3.01625        $ bottom horizontal OD
99 rpp  -250 250 -250 250 -250 250                  $ outside world

c
c data cards
c
c ****************************************************************
c Water Density = 1.0 g/cm^3
c Composition by atom fraction
c ****************************************************************
c
m1    1001 2.0000 $ H
      8016 1.0000 $ O
c
c ****************************************************************
c Stainless Steel 304
c Density = 7.94 g/cm^3 SCALE Standard Comp. Library
c ****************************************************************
c
m2    26000 -0.68375 $ Fe
      24000 -0.19000 $ Cr
      28000 -0.09500 $ Ni
      25055 -0.02000 $ Mn
      14000 -0.01000 $ Si
       6012 -0.00080 $ C
      15031 -0.00045 $ P
c
c **************************************************************
c AIR: ANSI/ANS-6.6.1, Dry air; density = 0.001225 g/cm^3
c Composition by weight fraction
c ***************************************************************
c
m3     7014 -0.75519 $ N
       8016 -0.23179 $ O
       6012 -0.00014 $ C
      18000 -0.01288 $ Ar
c
c **************************************************************
c AIR: ANSI/ANS-6.6.1, Dry air; density = 0.001225 g/cm^3
c Composition by weight fraction
c ***************************************************************
c this for detector plotting
m4     7014 -0.75519 $ N
       8016 -0.23179 $ O
       6012 -0.00014 $ C
      18000 -0.01288 $ Ar
c
c
mode  p
c
sdef   cel=d1 rad=d2 ext=fcel=d3 pos=fcel=d4 axs=fcel=d5
       erg=d6
c
si1 l  2 4 6 8
sp1 d  4.30102E+03  4.69983E+03 8.23132E+02 1.97192E+03 $ MCNP volumes
si2    2.6347
ds3 s  31 32 33 34
si31   97.627
si32   106.67
si33   18.684
si34   44.76
ds4 s  41 42 43 44
si41 l -61.30544 0 25.24125
```

```
sp41 d  1
si42 l  36.322 0 -84.455
sp42 d  1
si43 l  61.0385 0 -81.43875
sp43 d  1
si44 l  87.1145 0 -191.136
sp44 d  1
ds5 s  51 52 53 54
si51 l  1 0 0
sp51 d  1
si52 l  0 0 1
sp52 d  1
si53 l  1 0 0
sp53 d  1
si54 l  1 0 0
sp54 d  1
#     si6         sp6   sb6 $ this will create a response function by group
      1.0000E-02  0     0
      5.0000E-02  1     50
      1.0000E-01  1     1
      2.0000E-01  1     1
      3.0000E-01  1     1
      4.0000E-01  1     1
      6.0000E-01  1     1
      8.0000E-01  1     1
      1.0000E+00  1     1
      1.3300E+00  1     1
      1.6600E+00  1     1
      2.0000E+00  1     1
      2.5000E+00  1     1
      3.0000E+00  1     1
      4.0000E+00  1     1
      5.0000E+00  1     1
      6.5000E+00  1     1
      8.0000E+00  1     1
      1.0000E+01  1     1
c
fc4   R-9 DOSE RATE TALLY
f4:p  1
fm4   2.12E+05     $ 1 pho/sec per group; 18 groups; x source volume
ft4   scx 6  $ tally contribution by group
c flux-to-dose in rem/hr
de0 log  0.01 0.03 0.05 0.07 0.10 0.15
     0.20 0.25 0.30 0.35 0.40 0.45
     0.50 0.55 0.60 0.65 0.70 0.80
     1.00 1.40 1.80 2.20 2.60 2.80
     3.25 3.75 4.25 4.75 5.00 5.25
     5.75 6.25 6.75 7.50 9.00 11.00
     13.00 15.00
df0 log  3.96E-06 5.82E-07 2.90E-07 2.58E-07 2.83E-07 3.79E-07
     5.01E-07 6.31E-07 7.59E-07 8.78E-07 9.85E-07 1.08E-06
     1.17E-06 1.27E-06 1.36E-06 1.44E-06 1.52E-06 1.68E-06
     1.98E-06 2.51E-06 2.99E-06 3.42E-06 3.82E-06 4.01E-06
     4.41E-06 4.83E-06 5.23E-06 5.60E-06 5.80E-06 6.01E-06
     6.37E-06 6.74E-06 7.11E-06 7.66E-06 8.77E-06 1.03E-05
     1.18E-05 1.33E-05
c
ctme  20
```

## 13 APPENDIX: NEI 99-01 REV 5 EXCERPTS

**SYSTEM MALFUNCTIONS**

**SU4**

**Initiating Condition - NOTIFICATION OF UNUSUAL EVENT**

Fuel Clad degradation.

**Operating Mode Applicability:** Power Operation, Startup, Hot Standby, Hot Shutdown

**Example Emergency Action Levels:** (1 or 2)

1. (Site specific radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits.)
2. (Site specific coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.)

**Basis:**

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the Fission Product Barriers.

EAL #1

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

[*Such as BWR air ejector monitors, PWR failed fuel monitors, etc.*]

EAL #2

This threshold addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits.

**Basis Information For**
**PWR EAL Fission Product Barrier Table 5-F-3**

## FUEL CLAD BARRIER THRESHOLDS: (1 or 2 or 3 or 4 or 6 or 7 or 8)

The Fuel Clad barrier consists of the zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.

### 1. Critical Safety Function Status

[*These thresholds are for PWRs using Critical Safety Function Status Tree (CSFST) monitoring and functional restoration procedures. For more information, please refer to Section 3.9 of this document.*]

Loss Threshold A

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

Potential Loss Threshold A

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur.

Potential Loss Threshold B

Heat Sink - RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.

### 2. Primary Coolant Activity Level

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

[*The value can be expressed either in mR/hr observed on the sample or as µCi/gm results from analysis.*]

There is no Potential Loss threshold associated with this item.

### 3. Core Exit Thermocouple Readings

[*Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked) or plants which do not have a CSF scheme.*]

Loss Threshold A

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

[*This value typically corresponds to the temperature reading that indicates core cooling - RED in Fuel Clad Barrier loss threshold 1.A which is usually about 1200 degrees F.*]

## 14 REFERENCE 6.14

WD001510

ENGINEERING CHANGE NOTICE

RGE-310

| COPIES TO: | | 1 | D. Beatty |
|---|---|---|---|
| 1 | W. Patalon ✓ | 1 | E. U. Powell |
| 1 | S. G. Caslake | 1 | E. J. Staffel |
| 1 | A. A. Simmons | 1 | File RGE-310/1.1 |

E-NCS-1007

SERIAL NO. ~~RGE-~~ ECO- 70070 | CLASS 1 | CLASS 2

DESCRIPTION OF CHANGE (INCLUDE NATURE, REASON, INSTRUCTIONS TO DRAFTING, TEST DATA, ETC.)

SUBJECT: FAILED FUEL ELEMENT DETECTOR

REFERENCES: (a) EUP-2150 dated April 23, ~~1960~~ 1969

Ref. (a) stated that a failed fuel detector in the form of a G-M tube on the letdown line is to be added to the RGE Plant at the request of the AEC-DRL. This failed fuel element detector is to be an area monitor channel (R-9).

The drawer will be mounted in the Radiation Monitoring Equipment rack by Tracerlab.

The detector is to be mounted by the field. The detector is to be located in line with and approximately six inches away from the letdown line inside the spray addative tank cubicle (i.e., between the non-regenerative heat exchanger and TCV 145). The alarm set point is $1 \times 10^3$ mr/Hr.

The required cabling is to be the same type (Tracerlab type A503247) as cables 9 through 16 listed in the cable legend shown on Tracerlab drawing D997498.

Tracerlab is supplying the mounting bracket for the detector.

The equipment is scheduled to be shipped from Tracerlab approximately September 1, 1969.

INADMISSABLE AS EVIDENCE PURSUANT TO THE AGREEMENT OF RG&E AND WESTINGHOUSE

Westinghouse Proprietary Class 2C

PART EFFECT

| | | |
|---|---|---|
| NONE | Estimated Change in Product Cost | Capital Equipment Required |
| SALVAGE | $ | Charge No. ~~RGE-330~~ |
| SCRAP | | Design Spec. No. |

Initiated By W. B. Kearton *W. Kearton* Date August 26, 1969

APPROVED BY:

| FUNCTIONAL ENG. & DATE | PROJECT ENG. & DATE | MFG. ENG. & DATE | COMPLETED BY DRAFTING |
|---|---|---|---|
| W. Patalon *CKP 8/28/69* | *for A. A. Simmons 8/28/69* | | |

**ATTACHMENT 1, DESIGN VERIFICATION REPORT (FOR INFORMATION ONLY)**

| | | | |
|---|---|---|---|
| **Doc Type being verified:** | ☐ Design Change ☒ Calculation ☐ Specification<br>☐ Other: | | |
| **Document No.:** | **CALC-2011-0019** | **Rev.:** | 000 |

**Extent of Design Verification** (Briefly describe):

Checked inputs, spreadsheet, ORIGEN and MCNP input and output.

**Method of Design Verification:**

☒ Design Review ☐ Qualification Testing
☐ Alternate Calculations ☐ Applicability of Proven Design

**Results of Design Verification:**

☐ Fully acceptable with no issues identified
☒ Fully acceptable based on the following issues identified and resolved:

| No. | Verifier's Comment | Preparer's Response (Indicate if not required) | Verifier Concurs? (Yes/No) |
|---|---|---|---|
| 1 | Table 4-1 – Cs-138 should be 1.06 uCi/g instead of 1.08 uCi/g per Table 2-1 of Reference 6-3. | Corrected. | Y |
| 2 | Input 4.4 – Provide a basis for the assumption of 35,000 MWd/MTU used to select gap fractions. Review of core loadings for cycles 34-36 indicates that BOC core average BU is ~21 GWd/MTU, MOC is ~30 GWd/MTU, and EOC is ~39.5 GWd/MTU (based on Table 4-1 in CN-REA-08-11, CN-REA-09-53, and CN-REA-11-7). | The basis was engineering judgment. The use of the gap fractions is explained in Assumption 5.2. A clarification of this burnup is added to 5.2 with respect to recent past cycle operating history. | Y |

| | | | |
|---|---|---|---|
| 3 | Assumption 5.1 and Section 10.2 – The R-9 detector was modeled as being 6 inches from both the horizontal and vertical runs of letdown piping. Visual inspection of Figure 10-2 suggests that the detector is farther from the vertical run than the horizontal run. The Figure 4-1 distances from the bottom end of the coupling on the vertical line to the center of the horizontal piping at El. 241.5 ft (1.68 ft) was used to scale Figure 10-2 to estimate distances between the detector and the piping. This estimate yielded ~7 inches from the detector to the horizontal piping and ~12 inches to the vertical piping. This would lower estimated dose rates, but may be compensated for by the lack of modeling of the concrete wall (neglects backscatter). The successful benchmark to UFSAR section 9.3.4.4.9.3 suggests these two effects may cancel each other out. | The visual inspection of Figure 10-2 for dimensions is difficult due to the parallax error present. The apparent 7 inch distance from the top horizontal letdown line to the detector is expected to be close to 6" without parallax. That said, review of the figure again does suggest the detector is farther than 6" from the vertical line. To compensate the detector was moved to 10" from the pipe. | |
| 4 | Assumption 5.2 – 0.1% failed fuel is equivalent to about 22 rods based on 121 assemblies per core and 179 rods per assembly. Also, the DEQ I-131 value of 1.6 uCi/g appears to be based on pre-EPU core inventories of iodine. Using the EPU core inventories indicated in DA-NS-2002-037 (used in all AST and EQ dose analyses for EPU) a DEQ I-131 of 2.0 uCi/g is calculated for 0.1% failed fuel. This may be appropriate if used only for the UFSAR benchmark case (if so you should indicate this explicitly where 1.6 DEQ I-131 is mentioned). | Agreed. However, the 1.6 uCi/gm case was intended to use pre-EPU sources since the UFSAR 200 mrem/hr is believed to be the setpoint prior to EPU. | Y |
| 5 | Assumption 5.3 – "man half-lives" should be "many half-lives" | Corrected. | Y |
| 6 | Section 5 – Should probably add an assumption that the insulation shown on the lower half of the vertical section of piping has negligible shielding value was ignored in the MCNP model. | Corrected. Added Assumption 5.6. | Y |
| 7 | Section 11.2 and 11.3 - The response function listed in Table 11-2 appears to use the relative dose rate multipliers based on the upper bin energy rather than the average of the upper and lower energies of the bin. This would yield conservative dose rates, but may not be appropriate for a calculation intended to be best estimate. However, this may also be a compensatory measure for some of the approximations noted in comment 3. | As shown in source distribution 6 the gamma energy spectrum was entered in histogram format. And the special treatment ft4 scx 6 bins the f4 tally by the source energy groups. Therefore the response function is group-wise, not by upper bin energy. | Y |

| | | | |
|---|---|---|---|
| Lead Design Verifier: | J. R. MASSARI | [signature] | 12/19/2011 |
| | Name | Signature | Date |

Engineering Manager (if required): N/A

Discipline Design Verifiers if required: N/A

| Discipline | Name | Signature | Date |
|---|---|---|---|
| | | | |
| | | | |
| | | | |

(USE ADDITIONAL SHEETS AS REQUIRED)

Check if Design Review Checklist is attached ☒

Check if additional sheets are used ☐

**ATTACHMENT 2, DESIGN VERIFICATION CHECKLIST (FOR INFORMATION ONLY)**

The following questions are required to be addressed based on Constellation Nuclear Generation commitment to ANSI/ASME NQA-1-1994 for design verification activities. This checklist is intended to assist when using the Design Review method of design verification to ensure relevant items are addressed in the verification effort. Each "No" answer will require correction or resolution by the originator of the document being verified prior to full acceptance by the design verifier(s).

Doc #: **CALC-2011-0019** Rev 000 Lead Design Verifier's Name: J. R. MASSARI

| | Review Check | | |
|---|---|---|---|
| | Yes | No | N/A |
| 1. Were the design inputs correctly selected? | X | | |
| 2. Are assumptions necessary to perform the design activity adequately described and reasonable? | X | | |
| Where necessary, are the assumptions identified for subsequent re-verifications when the detailed design activities are completed? | X | | |
| 3. Was an appropriate design method used? | X | | |
| 4. Were the design inputs correctly incorporated into the design? | X | | |
| 5. Is the design output reasonable compared to design inputs? | X | | |
| 6. Are the necessary design input and verification requirements for interfacing organizations specified in the design documents or in supporting procedures or instructions? | | | X |

# CALCULATION COVER SHEET

**A. INITIATION**



Site ☐ CCNPP ☐ NMP ☒ REG

Calculation No.: **CALC-2011-0020** Revision No.: 000

Vendor Calculation (Check one): ☐ Yes ☒ No

Responsible Group: SHANE R. GARDNER

Responsible Engineer: NUCLEAR ANALYSIS UNIT - FLEET NUCLEAR FUELS

**B. CALCULATION**

ENGINEERING DISCIPLINE: ☐ Civil ☐ Instr & Controls ☒ Nuc Eng ☐ Electrical ☐ Mechanical ☐ Nuc Fuel Mgmt ☐ Other: ☐ Reliability Eng

Title: NEI 99-01 TECHNICAL BASIS FOR THE GINNA EFFLUENT MONITOR EMERGENCY ACTION LEVELS (EALS)

Unit: ☒ 1 ☐ 2 ☐ COMMON

Proprietary or Safeguards Calculation: ☐ YES ☒ NO

Comments: SUPPORTS NEI 99-01 UPGRADE

Vendor Calc No.: N/A REVISION NO.: N/A

Vendor Name: N/A

Safety Class (Check one): ☐ SR ☐ AUGMENTED QUALITY ☒ NSR

There are assumptions that require Verification during walkdown: N/A TRACKING ID:

This calculation **SUPERSEDES**: N/A

**C. REVIEW AND APPROVAL:**

Responsible Engineer: S. R. GARDNER Shane R. Gardner 12/19/2011

Printed Name and Signature Date

Is Design Verification Required? ☐ Yes ☒ No

If yes, Design Verification Form is ☐ Attached ☐ Filed with:

Independent Reviewer: J. R. MASSARI [signature] 12/19/2011

Printed Name and Signature

Approval: PHILIP WENGLOSKI [signature] 12/20/11

Printed Name and Signature Date

## REVISION SUMMARY AND LIST OF EFFECTIVE PAGES

Initial Issue – All pages Rev. 0.

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## 1 OBJECTIVE AND PURPOSE

The objective of this calculation is to determine the Emergency Action Level (EAL) values for the effluent radiation monitors in P-9 (Reference 6.1) in accordance with the guidance provided in NEI 99-01 Revision 5 (Reference 6.2). The calculation will provide or summarize the EALs for the gaseous and effluent radiation monitors.

## 2 SCOPE OF CALCULATION

Future changes in the EAL technical bases will require review of this calculation for continued applicability or revision. This calculation does not establish Offsite Dose Calculation Manual (ODCM) release limits and any modification of ODCM limits may necessitate revision of this calculation.

## 3 CONCLUSIONS

The gaseous effluent monitor EAL values based on NEI 99-01 Rev. 5 are reported to two significant figures in Table 3-1.

**TABLE 3-1 EFFLUENT RADIATION MONITOR EAL VALUES BASED ON NEI 99-01 REV. 5**

| Release | Monitor | GE | SAE | ALERT | UE | Note |
|---|---|---|---|---|---|---|
| CNMT | R-12 | N/A | N/A | N/A | 7.4E+6 (cpm) | 1 fan operation |
| | R-12 | N/A | N/A | N/A | 5.1E+6 (cpm) | 2 fan operation |
| | R-12A | 1.8E+2 (uCi/cc) | 1.8E+1 (uCi/cc) | 1.8E+0 (uCi/cc) | N/A | |
| Plant Vent | R-14 | N/A | N/A | N/A | 6.0E+5 (cpm) | |
| | R-14A | 2.1E+1 (uCi/cc) | 2.1E+0 (uCi/cc) | 2.1E-1 (uCi/cc) | N/A | |
| Air Ejector | R-15 | N/A | N/A | N/A | 6.3E+5 (cpm) | |
| | R-48 | 5.7E+2 (uCi/cc) | 5.7E+1 (uCi/cc) | 5..7E+0 (uCi/cc) | N/A | |
| 1 ARV | R-31/R-32 | 5.0E+3 (mR/hr) | 5.0E+2 (mR/hr) | 5.0E+1 (mR/hr) | 8.0E+0 (mR/hr) | |
| 1 Safety | | 2.3E+3 (mR/hr) | 2.3E+2 (mR/hr) | 2.3E+1 (mR/hr) | 3.7E+0 (mR/hr) | |
| 2 Safeties | | 1.1E+3 (mR/hr) | 1.1E+2 (mR/hr) | 1.1E+1 (mR/hr) | N/A | |
| 3 Safeties | | 7.7E+2 (mR/hr) | 7.7E+1 (mR/hr) | 7.7E+0 (mR/hr) | N/A | |
| 4 Safeties | | 5.7E+2 (mR/hr) | 5.7E+1 (mR/hr) | 5.7E+0 (mR/hr) | N/A | |

| Release | Monitor | GE | SAE | ALERT | UE |
|---|---|---|---|---|---|
| Liq Radwaste | R-18 | N/A | N/A | N/A | 3.6E+5 (cpm) |
| SFP HX A | R-20A | N/A | N/A | N/A | 4.0E+4 (cpm) |
| SFP HX B | R-20B | N/A | N/A | N/A | 5.2E+3 (cpm) |
| TB Floor Drains | R-21 | N/A | N/A | N/A | 5.0E+4 (cpm) |
| Hi Cond Waste | R-22 | N/A | N/A | N/A | 9.2E+4 (cpm) |

## 4 DESIGN INPUTS

4.1. The key inputs to the determination of the gaseous effluent EAL values are shown in Table 4-1 below. The other key inputs are the air immersion Total Effective Dose Equivalent (TEDE) dose conversion factors (DCFs). Since the TEDE DCF is dependent on the mixture of radionuclides it is developed in Assumption 5.2.

4.2. The key inputs to the determination of the liquid EAL values are the ODCM release rate limits. The precaution and limitation Section 5.2 of P-9 (Reference 6.1) states that the release rate limits are determined in accordance with the methods of the ODCM; i.e. the release limits listed in P-9 are the ODCM release limits. Therefore, for the purposes of this calculation the ODCM limits come from P-9, unless specified otherwise.

4.3. The NEI 99-01 Rev. 5 sets the initiating conditions for radiological effluent EALs. These are covered in the following EALS: AG1 General Emergency (GE), AS1 Site Area Emergency (SAE), AA1 Alert and AU1 Unusual Event (UE). The fundamental characteristics (see Table 5-A-1 of NEI 99-01) associated with these EALs are as follows:

a) GE – 1,000 mrem TEDE or 5,000 mrem Thyroid over 1 hour

b) SAE – 100 mrem TEDE or 500 mrem Thyroid over 1 hour

c) Alert – 200 x ODCM limits (noble gas)

d) UE – 2 x ODCM limits (noble gas)

e) Meteorology should be consistent with ODCM derived values. Therefore, the ODCM annual average X/Q values for the effluent monitors will be used (see Table 2-2 of ODCM).

f) The gaseous dose limit associated with ODCM is 500 mrem/yr whole body per ODCM Section 2.6. Therefore the UE would be 2 x 500 mrem/yr x 1 yr/8760 hrs = 0.1 mrem/hr. The Alert is 100x the UE, or 10 mrem/hr. See Appendix A, page 156, of NEI 99-01 for an explanation. Therefore, the effective transition points for the gaseous EALs are:

   i) 1,000 mrem/hr General Emergency

   ii) 100 mrem/hr Site Area Emergency

   iii) 10 mrem/hr Alert

   iv) 0.1 mrem/hr Unusual Event

g) The dose criteria above apply to a receptor at or beyond the site boundary.

4.4. The ODCM Xe-133 immersion dose conversion factor for release limit setpoint calculations is 294 mrem/yr per uCi/$m^3$ (see Table 2-3 of the ODCM). This value is converted to rem/hr per uCi/cc by multiplying by (1E6 cc/$m^3$)/(8760 hrs/yr) or 1/8.76. The converted value is 33.6 rem/hr per uCi/cc.

## TABLE 4-1 GASEOUS EFFLUENT RADIATION MONITOR EAL INPUTS

| Release | Monitor | Flow Rate (cfm) | Reference | X/Q (sec/m3) | Reference |
|---|---|---|---|---|---|
| CNMT | R-12A | 15,300 | EPIP-2-4 | 1.60E-06 | ODCM Table 2-2 |
| Plant Vent | R-14A | 77,000 | Assumption | 2.70E-06 | ODCM Table 2-2 |
| Air Ejector | R-15 | 600 | DA-RP-2001-018 | 1.30E-05 | ODCM Table 2-2 |
| | R-48 | | | 6.50E-05 | |
| ARV | R-31/R-32 | 2,565 | EPIP-2-3 | 2.70E-06 | Assumption |
| Safety | R-31/R-32 | 5,550 | EPIP-2-3 | 2.70E-06 | Assumption |

| Release | Detector | ODCM Limit | Units | Reference |
|---|---|---|---|---|
| CNMT 1 fan | R-12 | 3.71E+06 | cpm | DA-RP-98-091 |
| CNMT 2 fan | R-12 | 2.56E+06 | cpm | DA-RP-98-091 |
| Plant Vent | R-14 | 3.00E+05 | cpm | DA-RP-98-091 |
| Air Ejector | R-15 | 3.20E+05 | cpm | DA-RP-2001-019 |
| MS - 1 ARV | R-31/R-32 | 0.1[1] | mR/hr | P-9 |
| MS - 1 Safety | R-31/R-32 | 0.1[1] | mR/hr | P-9 |
| Liq Radwaste | R-18 | 1.80E+05 | cpm | P-9 |
| SFP HX A | R-20A | 2.04E+04 | cpm | P-9 |
| SFP HX B | R-20B | 2.60E+03 | cpm | P-9 |
| TB Floor Drains | R-21 | 2.50E+04 | cpm | P-9 |
| Hi Cond Waste | R-22 | 4.60E+04 | cpm | P-9 |

1. Note, the MS relief valve pathways are not called out as an effluent pathway. The P-9 states the release limit corresponds to a monitor setpoint just above background. Since this is not set by the ODCM, a new release rate limit will be calculated.

| Release | Monitor | Detector Calibration Factor (uCi/cc/cpm) | Reference |
|---|---|---|---|
| CNMT | R-12 | 5.50E-08 | DA-RP-98-091 |
| Plant Vent | R-14 | 5.60E-08 | DA-RP-98-091 |
| Air Ejector | R-15 | 1.45E-06 | DA-RP-2001-019 |

## 5 ASSUMPTIONS

5.1. A dose rate conversion factor of 0.1295 uCi/cc per mrem/hr (at t=0) is assumed for converting the calculated main steam safety/relief valve effluent radiation monitor (R-31/R-32) values. This value is justified by the assumption that the source terms are not decayed. This value comes from EPIP-2-3 Attachment 3 (Reference 6.11)

5.2. It is assumed that the air immersion TEDE dose conversion factor of 471.74 rem/hr per uCi/cc, with no decay, is applicable to gaseous effluent monitor calculations. This value is chosen to be consistent with the Emergency Planning (EP) dose projection software. It is also based on the EPA Protective Action Guidelines (PAGs) (Reference 6.14), which is consistent with NEI 99-01 Rev. 5. A basis for using this factor for the Alert and above calculations is developed in Appendix Section 12.

5.3. It is assumed that the air immersion TEDE dose conversion factor of 42.3 rem/hr per uCi/cc, with no decay, is representative of main steam safety/relief discharges without significant fuel failure or steam generator tube leaks. This value is based on the activity concentrations in the RCS and the basis is developed in Appendix Section 12.

5.4. It is assumed that the liquid effluent release pathway is not capable of reaching a dose level of 10 mrem/hr (Alert EAL) at the nearest potable water supply. This is supported by the fact that liquid discharges are greatly reduced by dilution and that liquid releases pathways are isolated upon high radiation signals. Therefore, the GE, SAE and Alert EALs are not applicable to the liquid effluent radiation monitors.

5.5. For R-48 calculations the ODCM X/Q for R-15 is used for consistency. This is a good assumption because the R-15 X/Q value is the bounding land sector whereas the R-48 X/Q corresponds to a worst sector over the lake. The R-15 X/Q is more appropriate for EP.

5.6. For the main steam safety/relief valve release the X/Q of the plant vent is assumed. This assumption is taken as there is no X/Q listed in the ODCM for this release. This assumption is justified by the similarity in the release characteristics between the plant vent and the main steam releases. Per drawing 33013-1231 (Reference 6.19), the ARV stack diameter is 12" and the safeties stack diameter is 14". The plant vent is 54" diameter. Using the flow rates from Table 4-1 and the stack area the flow velocities were compared. The main steam safeties and the plant vent effluent velocity are on the order of 80 ft/sec. The main steam ARVs are slightly less on the order of 50 ft/sec. The release points are horizontally located in the same roof area (intermediate building) of the plant (Reference 6.20). However, there is an approximate 40 ft difference in elevation (Reference 6.21). Since the ARV/safeties are sheltered by the façade of the containment building this elevation difference is minimized. Therefore, it is reasonable to assume the ODCM plant vent X/Q for the main steam releases.

5.7. The plant vent flow rate is assumed to be 77,000 cfm for all EAL conditions above UE. This assumption is made to cover the range of values in the reference documentation. DA-RP-2001-017 (Reference 6.18) considers the design flow rate of the system to be 75,000 cfm, but states that flow surveillances verify the flow is greater than 80,000 cfm (also shown in P-9). The value in EPIP-2-4 (page 4) is 77,844 cfm normal and 71,289 cfm emergency, while the value in EPIP-2-3 (Attachment 1) is 76,000 cfm. Based on the above, 77,000 cfm is a reasonable assumption for EAL calculations.

# 6 REFERENCES

6.1 P-9, Rev. 09807, "Radiation Monitoring System."

6.2 NEI 99-01, Rev. 5, "Methodology for Development of Emergency Action Levels."

6.3 ODCM, Revision 27.

6.4 CN-REA-04-34, Rev. 0, "R. E. Ginna RCS Sources for 1811 MWt Uprate."

6.5 CN-REA-04-57, Rev. 1, "R. E. Ginna Normal Operation RCS and Secondary Coolant Sources for 1811 MWt Uprate."

6.6 CN-REA-04-32, Rev. 0, "Ginna 1811 MW Uprate – Core, Fuel Handling Accident, and POST/FIPCO ORIGEN Sources."

6.7 CN-REA-02-33, Rev. 0, "Radiation Source Terms for R. E. Ginna Alternate Source Term Analysis."

6.8 DA-RP-98-091, Rev. 3, "Radiation Process Monitors Setpoint Calculations."

6.9 DA-RP-2001-018, Rev. 2, "Air Ejector Process Radiation Accident Monitor R-47 & R-48 Setpoints."

6.10 DA-RP-2001-019, Rev. 2, "Air Ejector Process Radiation Monitor R-15 Setpoints."

6.11 EPIP-2-3, Rev. 01800, "Emergency Release Rate Determination."

6.12 EPIP-2-4, Rev. 01700, "Emergency Dose Projections – Manual Method."

6.13 DA-NS-2001-087, Rev. 4, "Large-Break LOCA Offsite and Control Room Doses."

6.14 EPA-400, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," May 1992.

6.15 USNRC NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," Dec. 1997.

6.16 USNRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

6.17 DA-RP-08-064, Rev. 0000, "Technical Basis of the Ginna Nuclear Power Plant Dose Projection Program Calculations."

6.18 DA-RP-2001-017, Rev. 2, "Plant Vent Radiation Accident Monitor RM-14A Setpoints."

6.19 33013-1213, Rev. 40, "Main Steam (MS) P & ID."

6.20 33013-2122, Rev. 2, "Plant Arrangement Containment Struct. Tendon Access Platform & Main Steam Access Plat."

6.21 33013-2131, Rev. 1, "Plant Arrangement Reactor Containment Structure Section 1-1."

## 7 DOCUMENTATION OF COMPUTER CODES

RADTRAD (Reference 6.15) was used in this calculation. Spreadsheets were used for input preparation and data reduction.

Computer files associated with this calculation are listed in Table 7-1. The files are stored in electronic format in the configuration management system. No CD-ROMs are part of this calculation.

**TABLE 7-1 LISTING OF COMPUTER FILES**

| | | | |
|---|---|---|---|
| 12/19/2011 | 11:41 AM | 110,818 | CALC-2011-0020 REV000.xlsx |
| 12/13/2011 | 12:42 PM | 241,904 | contleak.o0 |
| 12/13/2011 | 12:42 PM | 5,228 | contleak.psf |
| 12/13/2011 | 01:21 PM | 79,848 | eccs.o0 |
| 12/13/2011 | 01:21 PM | 4,219 | eccs.psf |
| 09/10/2010 | 01:55 PM | 978 | ECCS.RFT |
| 12/02/2009 | 01:20 PM | 9,352 | Ginna LOCA63.nif |
| 12/01/2009 | 01:31 PM | 51,396 | Ginna.inp |
| 12/03/2009 | 03:17 PM | 983 | LOCA.RFT |
| 08/29/2002 | 08:07 AM | 824 | PWR_DBA.RFT |

## 8 METHOD OF ANALYSIS

To determine the gaseous EAL values a dose calculation is performed for an airborne release. The dose rate is calculated using the following equation:

**EQUATION 8-1**

D = R * X/Q * DCF,

where D is the dose rate, R is the activity release rate, X/Q is the atmospheric dispersion coefficient and DCF is the nuclide-specific airborne dose conversion factor.

The activity release rate is defined by the product of the activity concentration, A, and the volumetric flow rate, F:

**EQUATION 8-2**

R = A * F

Substituting the above and re-arranging gives:

**EQUATION 8-3**

A = D / (CF * F * X/Q * DCF),

where,

A = the activity concentration in uCi/cc

F = *the flow rate in cfm*

X/Q = the atmospheric dispersion factor in sec/$m^3$

DCF = the air immersion TEDE dose conversion factor in rem/hr per uCi/cc

and,

CF = is a conversion factor = 1 min / 60 sec * $(0.3048\ m)^3$ / 1 $ft^3$ = (1/2118.88) min-$m^3$/sec-$ft^3$.

Equation 8-3 is used to calculate the activity concentration that corresponds to the specified dose rate, D, which is set by the EAL.

If the radiation monitor system does not output activity concentration then it can be converted by a response factor, RF, to achieve the desired monitor response units. This typically is either a dose rate or count rate factor. The calculation to determine the monitor response from the calculated activity concentration is shown by the equation below:

**EQUATION 8-4**

EAL = A * RF,

where, RF is the response factor in units of (x) per uCi/cc; (x) is the monitor output units.

The above equations are used to calculate the EAL for each initiating condition. For initiating conditions that are driven by ODCM the methodology prescribed by the NEI 99-01 Rev. 5 is used.

## 9 ACCEPTANCE CRITERIA

There are no acceptance criteria for this calculation. The results are used to define the technical bases for EALs in the Emergency Plan.

## 10 CALCULATIONS/COMPUTATIONS

The calculations are performed in the attached spreadsheet CALC-2011-0020 REV000.xlsx. These are hand calculations and can easily be examined by inspecting the formulas. The variables identified in the methodology are identified in the heading columns. The gaseous monitor calculations can be found in the EAL CALC – FINAL worksheet. The liquid monitor calculations can be found in EAL – LIQUID worksheet.

As identified in Table 4-1 the release rate limit for the main steam relief/safety valves pathway in P-9 is not derived from the ODCM. Therefore, it is necessary to calculate a value. For this calculation the ODCM dose limit of 500 mrem/yr is used and the Xe-133 dose conversion factor from Design Input 4.4 is used. The resulting R-31/R-32 monitor response is the value corresponding to the requirements of ODCM. Since the P-9 limit on R-31/R-32 is tied to background the ODCM limit was much higher. This value is then used to calculate the UE EAL value.

Additional calculations were performed to assess the EAL transitions. This was done because of the potential gap that can occur with the transition from the ODCM EAL to the higher level EAL (GE, SAE or Alert). In this calculation, this transition occurs in going from the UE to the SAE. The reason a gap can occur is because of the underlying bases behind the dose calculation. In ODCM the setpoints are determined for Xe-133 using Regulatory Guide 1.109 (Reference 6.16) dose conversion factors. In all gaseous calculations the ODCM X/Q values are used thus eliminating the potential for a gap. To assess the transition, the ALERT value is compared to the ODCM UE EAL. The ODCM UE should be less than the Alert EAL, by the factor of 100. To compare values of the same quantity, detector calibration factors are used to convert UE EALs from cpm to uCi/cc. The results of these calculations are shown in Table 10-1 and confirm that there is no overlap.

The results do show the differences in ODCM and the methodology used for the Alert EAL. The dose conversion factor causes a difference of a factor of 471.73 to 33.5 or about 14. When accounting for this difference the adjusted ALERT/UE ratio gets very close to about 100.

**TABLE 10-1 EAL TRANSITION COMPARISON**

| | | Calibration Factor | UE EAL Overlap Check with ODCM (uCi/cc) | | |
|---|---|---|---|---|---|
| Release | Detector | (uCi/cc/cpm) | UE | ALERT | ALERT/UE Ratio |
| CNMT 1 fan | R-12 | 5.50E-08 | 4.08E-01 | 1.83E+00 | 4.5 |
| CNMT 2 fan | R-12 | 5.50E-08 | 2.82E-01 | 1.35E+00 | 4.8 |
| Plant Vent | R-14 | 5.60E-08 | 3.36E-02 | 2.16E-01 | 6.4 |
| Air Ejector | R-15 | 1.45E-06 | 9.25E-01 | 5.76E+00 | 6.2 |
| MS - 1 ARV | R-31/R-32 | n/a | 1.04E+00 | 6.49E+00 | 6.2 |
| MS - 1 Safety | R-31/R-32 | n/a | 4.81E-01 | 3.00E+00 | 6.2 |

## 11 RESULTS

The following tables summarize the results of this calculation. Table 11-1 shows the inputs and calculation for determining the gaseous monitor EALs for the GE. The ODCM based gaseous monitor

EAL calculation results are shown in Table 11-2. Finally, the ODCM based liquid monitor EAL calculation results are shown in Table 11-3.

The results shown in the above described tables are combined in Table 11-4. The General Emergency EALs (1 rem/hr) are scaled by the appropriate factor of 10 to obtain the Site Area Emergency and Alert EALs. The Unusual Event EALs are taken from the 2x ODCM results. All results are adjusted to be rounded down to two significant digits.

**TABLE 11-1 GASEOUS EAL CALCULATION RESULTS**

| | | D | F | X/Q | DCF | A | RF | | EAL | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | Dose Rate Limit | Effluent Flow Rate | Atmospheric Dispersion Factor | TEDE Dose Conversion Factor | Concentration Limit | Monitor Calibration Factor | | | |
| | | | | | | | | Units | | |
| Release | Detector | (rem/hr) | (cfm) | (sec/m3) | (rem/hr per uCi/cc) | (uCi/cc) | Value | (x) per uCi/cc | Monitor Response Limit | |
| CNMT | R-12A | 1 | 15,300 | 1.60E-06 | 471.73 | 183.5 | 1 | N/A | 183.49 | (uCi/cc) |
| Plant Vent | R-14A | 1 | 77,000 | 2.70E-06 | 471.73 | 21.61 | 1 | N/A | 21.61 | (uCi/cc) |
| Air Ejector | R-48 | 1 | 600 | 1.30E-05 | 471.73 | 576 | 1 | N/A | 576 | (uCi/cc) |
| MS - 1 ARV | R-31/R-32 | 1 | 2,565 | 2.70E-06 | 471.73 | 648.6 | 7.722 | (mR/hr) | 5008 | (mR/hr) |
| MS - 1 Safety | R-31/R-32 | 1 | 5,550 | 2.70E-06 | 471.73 | 299.7 | 7.722 | (mR/hr) | 2315 | (mR/hr) |
| MS - 1 ARV | R-31/R-32 | 5.7078E-05 | 2,565 | 2.70E-06 | 33.56 | 5.2E-01 | 7.722 | (mR/hr) | 4.02E+00 | (mR/hr) |
| MS - 1 Safety | R-31/R-32 | 5.7078E-05 | 5,550 | 2.70E-06 | 33.56 | 2.4E-01 | 7.722 | (mR/hr) | 1.86E+00 | (mR/hr) |

**TABLE 11-2 ODCM DERIVED GASEOUS EAL RESULTS**

| Release | Detector | ODCM Limit | Units | x2 | x200 |
|---|---|---|---|---|---|
| CNMT 1 fan | R-12 | 3.71E+06 | (cpm) | 7.42E+06 | 7.42E+08 |
| CNMT 2 fan | R-12 | 2.56E+06 | (cpm) | 5.12E+06 | 5.12E+08 |
| Plant Vent | R-14 | 3.00E+05 | (cpm) | 6.00E+05 | 6.00E+07 |
| Air Ejector | R-15 | 3.20E+05 | (cpm) | 6.40E+05 | 6.40E+07 |
| MS - 1 ARV | R-31/R-32 | 4.02 | (mR/hr) | 8.04E+00 | 8.04E+02 |
| MS - 1 Safety | R-31/R-32 | 1.86 | (mR/hr) | 3.71E+00 | 3.71E+02 |

## TABLE 11-3 ODCM DERIVED LIQUID EAL RESULTS

| Release | Detector | ODCM Limit | Units | x2 | x200 |
|---|---|---|---|---|---|
| Liq Radwaste | R-18 | 1.80E+05 | (cpm) | 3.60E+05 | 3.60E+07 |
| SFP HX A | R-20A | 2.04E+04 | (cpm) | 4.08E+04 | 4.08E+06 |
| SFP HX B | R-20B | 2.60E+03 | (cpm) | 5.20E+03 | 5.20E+05 |
| TB Floor Drains | R-21 | 2.50E+04 | (cpm) | 5.00E+04 | 5.00E+06 |
| Hi Cond Waste | R-22 | 4.60E+04 | (cpm) | 9.20E+04 | 9.20E+06 |

## TABLE 11-4 EAL SUMMARY RESULTS SUMMARY TABLE

| Release | Monitor | GE | SAE | ALERT | UE | Note |
|---|---|---|---|---|---|---|
| CNMT | R-12 | N/A | N/A | N/A | 7.4E+6 (cpm) | 1 fan operation |
| | R-12 | N/A | N/A | N/A | 5.1E+6 (cpm) | 2 fan operation |
| | R-12A | 1.8E+2 (uCi/cc) | 1.8E+1 (uCi/cc) | 1.8E+0 (uCi/cc) | N/A | |
| Plant Vent | R-14 | N/A | N/A | N/A | 6.0E+5 (cpm) | |
| | R-14A | 2.1E+1 (uCi/cc) | 2.1E+0 (uCi/cc) | 2.1E-1 (uCi/cc) | N/A | |
| Air Ejector | R-15 | N/A | N/A | N/A | 6.3E+5 (cpm) | |
| | R-48 | 5.7E+2 (uCi/cc) | 5.7E+1 (uCi/cc) | 5.7E+0 (uCi/cc) | N/A | |
| 1 ARV | R-31/R-32 | 5.0E+3 (mR/hr) | 5.0E+2 (mR/hr) | 5.0E+1 (mR/hr) | 8.0E+0 (mR/hr) | |
| 1 Safety | | 2.3E+3 (mR/hr) | 2.3E+2 (mR/hr) | 2.3E+1 (mR/hr) | 3.7E+0 (mR/hr) | |
| 2 Safeties | | 1.1E+3 (mR/hr) | 1.1E+2 (mR/hr) | 1.1E+1 (mR/hr) | N/A | |
| 3 Safeties | | 7.7E+2 (mR/hr) | 7.7E+1 (mR/hr) | 7.7E+0 (mR/hr) | N/A | |
| 4 Safeties | | 5.7E+2 (mR/hr) | 5.7E+1 (mR/hr) | 5.7E+0 (mR/hr) | N/A | |

| Release | Monitor | GE | SAE | ALERT | UE |
|---|---|---|---|---|---|
| Liq Radwaste | R-18 | N/A | N/A | N/A | 3.6E+5 (cpm) |
| SFP HX A | R-20A | N/A | N/A | N/A | 4.0E+4 (cpm) |
| SFP HX B | R-20B | N/A | N/A | N/A | 5.2E+3 (cpm) |
| TB Floor Drains | R-21 | N/A | N/A | N/A | 5.0E+4 (cpm) |
| Hi Cond Waste | R-22 | N/A | N/A | N/A | 9.2E+4 (cpm) |

## 12 APPENDIX: AIR IMMERSION TEDE DOSE CONVERSION FACTOR BASIS

This appendix describes the basis for the TEDE dose factors assumed in the calculation of gaseous effluent monitor responses. Calculation DA-RP-08-064 (Reference 6.17) documents the basis for dose conversion factors (DCFs) for use in the emergency plan. They are derived from the EPA Protective Action Guidelines (PAGs) from EPA-400 (Reference 6.14). This is consistent with the approach in NEI 99-03 Rev. 5 (Reference 6.2).

In order to determine a single DCF the isotopic composition of the effluent during an accident must be utilized. In order to do this, DA-RP-08-064 assumes a generic LWR source term and common core release fractions from the basis of the Alternative Source Term (AST). These assumptions are valid for accidental source terms, however this appendix seeks to benchmark those assumptions for the use in EALs. Also, this appendix investigates the nobel gas DCF for RCS source terms in the primary and secondary side.

In order to benchmark the noble gas Effective Dose Equivalent (EDE) DCF after an accident a RADTRAD (Reference 6.15) model of CNMT leakage during a LOCA is used. The RADTRAD model is based on the LOCA calculations in DA-NS-2001-087 (Reference 6.13). The nuclide inventory released to the environment is extracted over time and is converted to a release rate. The isotopic mix is used to calculate the noble gas EDE and compare to DA-RP-080-064. The method of calculating the EDE is shown on page 9 of DA-RP-08-064 and is simply the source-weighted average EDE of the noble gas isotopes.

The RADTRAD model is shown in Section 12.1 below. The EDE dose conversion factors used are shown in Table 12-2 and are the same as those used in Attachment 4 of DA-RP-08-064. The relative composition of the noble gases derived from the RADTRAD results are shown in Table 12-1. Table 12-1 also shows the total EDE DCF. These results are plotted in Figure 12-1 and confirm the reasonableness of the DA-RP-08-064 noble gas EDE DCFs. These calculations are in the LOCA DCF Study worksheet of the CALC-2011-0020 REV000.xlsx spreadsheet.

The DCF used in the EAL calculations is based on the t=0 value from DA-RP-08-064. This is deemed appropriate as it conservatively represents the noble gas mix at the start of the accident. However for other events that don't involve large core damage the noble gas EDE DCF may be different. The RCS noble gases are evaluated to determine the impact on the DCF.

The design basis EPU RCS primary activities are based on CN-REA-04-34 (Reference 6.4) and the secondary steam activities are based on CN-REA-04-57 (Reference 6.5). The results of the calculation are shown in Table 12-3. As can be seen the noble gas EDE DCF is much lower than the t=0 post-LOCA value.

The t=0 noble gas EDE DCF in EPIP-2-4 (Reference 6.12) Attachment 3 is 470 rem/hr per uCi/cc. This is combined with the Committed Effective Dose Equivalent (CEDE) to obtain the TEDE. A noble gas to iodine ratio of 10,000:1 is assumed for this purpose. This is the default value in the Ginna EP dose projection methods and was challenged in NRC inspection report 50-244/89-19 dated 8/4/89. In the report, unresolved issue 50-244/88-14-08 was resolved and it was agreed that the 10,000:1 ratio was appropriate until the actual ratio could be determined from sample analysis. Therefore, the TEDE is calculated as follows:

TEDE = EDE + 1E-4*CEDE = 470 rem/hr per uCi/cc + 1E-4*(1.73E4 rem/hr per uCi/cc)

= 471.73 rem/hr per uCi/cc

where, the CEDE value is taken from DA-RP-08-064.

Based on the above assessment the TEDE DCF of 471.73 rem/hr per uCi/cc is a good assumption for EAL assessments. For assessment of events of lower consequence involving a release of noble gases a TEDE DCF of 42.3 rem/hr per uCi/cc is a good assumption based on RCS activity.

## TABLE 12-1 NOBLE GAS EFFECTIVE DOSE EQUIVALENT AS A FUNCTION OF TIME POST-ACCIDENT

| Nuclide | Relative Distribution of Noble Gas in CNMT Effluent Post-LOCA | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | **0.0139** | **0.0167** | **0.0194** | **0.022** | **0.5083** | **0.87** | **1.8083** | **2** | **8** | **24** | **96** | **720** |
| Kr-85m | 4.42E-02 | 4.44E-02 | 4.45E-02 | 4.47E-02 | 5.64E-02 | 6.11E-02 | 5.80E-02 | 5.50E-02 | 4.07E-02 | 1.16E-02 | 3.29E-04 | 0.00E+00 |
| Kr-85 | 1.91E-03 | 1.92E-03 | 1.92E-03 | 1.93E-03 | 2.63E-03 | 3.19E-03 | 3.57E-03 | 3.76E-03 | 4.37E-03 | 5.56E-03 | 7.71E-03 | 3.30E-02 |
| Kr-87 | 8.44E-02 | 8.46E-02 | 8.48E-02 | 8.49E-02 | 8.93E-02 | 7.27E-02 | 4.65E-02 | 3.34E-02 | 1.06E-02 | 1.99E-04 | 0.00E+00 | 0.00E+00 |
| Kr-88 | 1.19E-01 | 1.20E-01 | 1.20E-01 | 1.20E-01 | 1.46E-01 | 1.48E-01 | 1.28E-01 | 1.14E-01 | 6.72E-02 | 9.44E-03 | 6.00E-05 | 0.00E+00 |
| Xe-131m | 1.82E-03 | 1.83E-03 | 1.84E-03 | 1.85E-03 | 2.51E-03 | 3.04E-03 | 3.39E-03 | 3.57E-03 | 4.12E-03 | 5.10E-03 | 6.36E-03 | 1.29E-02 |
| Xe-133m | 1.03E-02 | 1.04E-02 | 1.04E-02 | 1.05E-02 | 1.41E-02 | 1.70E-02 | 1.88E-02 | 1.96E-02 | 2.19E-02 | 2.41E-02 | 1.93E-02 | 6.09E-03 |
| Xe-133 | 3.29E-01 | 3.31E-01 | 3.32E-01 | 3.33E-01 | 4.52E-01 | 5.48E-01 | 6.09E-01 | 6.39E-01 | 7.30E-01 | 8.74E-01 | 9.57E-01 | 9.48E-01 |
| Xe-135m | 6.25E-02 | 6.21E-02 | 6.16E-02 | 6.13E-02 | 2.61E-02 | 4.67E-03 | 5.17E-04 | 6.89E-05 | 3.29E-06 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| Xe-135 | 8.34E-02 | 8.37E-02 | 8.40E-02 | 8.43E-02 | 1.10E-01 | 1.27E-01 | 1.31E-01 | 1.31E-01 | 1.21E-01 | 7.00E-02 | 9.01E-03 | 1.77E-05 |
| Xe-138 | 2.63E-01 | 2.61E-01 | 2.58E-01 | 2.57E-01 | 1.01E-01 | 1.57E-02 | 1.51E-03 | 1.69E-04 | 7.24E-06 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| EDE | 424.5 | 423.5 | 422.7 | 421.9 | 345.2 | 279.0 | 230.5 | 205.0 | 132.0 | 45.5 | 26.3 | 30.1 |

EDE – units of rem/hr per uCi/cc

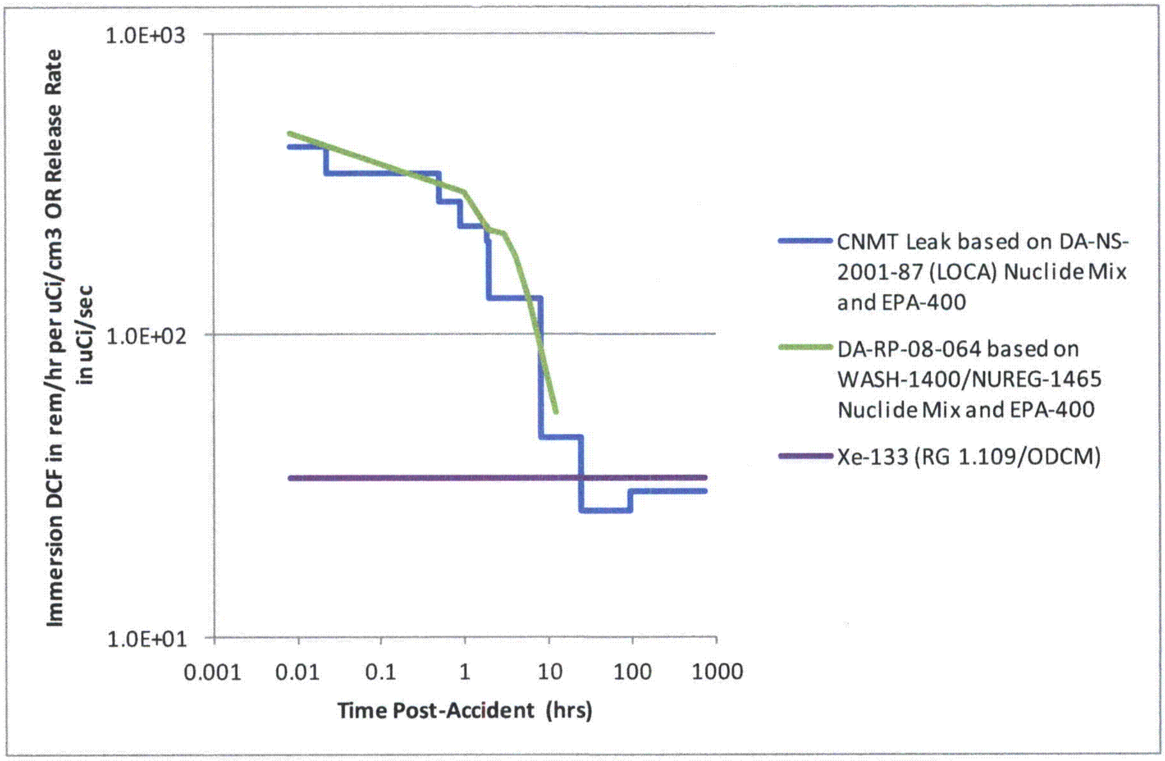
**TABLE 12-2 EPA-400 NOBLE GASE EFFECTIVE DOSE EQUIVALENT FACTORS**

| Nuclide | EDE (rem/hr per uCi/cc) |
|---|---|
| Kr-85m | 93 |
| Kr-85 | 1.3 |
| Kr-87 | 510 |
| Kr-88 | 1300 |
| Xe-131m | 850 |
| Xe-133m | 17 |
| Xe-133 | 20 |
| Xe-135m | 250 |
| Xe-135 | 140 |
| Xe-138 | 710 |

**TABLE 12-3 NOBLE GAS EFFECTIVE DOSE EQUIVALENT FACTORS FOR RCS RELEASES**

| Nuclide | EDE (rem/hr per uCi/cc) | RCS Activity (uCi/gm) Primary | RCS Activity (uCi/gm) Secondary (Steam) |
|---|---|---|---|
| Kr-85 | 1.3 | 8.21E+00 | 5.60E-07 |
| Kr-85m | 93 | 1.93E+00 | 6.50E-09 |
| Kr-87 | 510 | 1.24E+00 | 2.20E-08 |
| Kr-88 | 1300 | 3.60E+00 | 7.70E-09 |
| Xe-131m | 4.9 | 3.54E+00 | 3.20E-07 |
| Xe-133 | 20 | 2.71E+02 | 1.20E-08 |
| Xe-133m | 17 | 3.84E+00 | 2.80E-08 |
| Xe-135 | 140 | 9.49E+00 | 2.60E-08 |
| Xe-138 | 720 | 6.92E-01 | 3.10E-08 |
| Total | | 3.04E+02 | 1.01E-06 |
| EDE (rem/hr per uCi/cc) | | 42.3 | 50.1 |

**FIGURE 12-1 NOBLE GAS EFFECTIVE DOSE EQUIVALENT AS A FUNCTION OF TIME POST-LOCA**



## 12.1 RADTRAD LOCA CNMT MODEL INPUT FILES

**contleak.psf**

```
Radtrad 3.03 4/15/2001
 Cont Leak
 Nuclide Inventory File:
 c:\data\shane\work\ginna\rhr flyup issue sept 2010\radtrad\ginna loca63.nif
 Plant Power Level:
  1.8110E+03
 Compartments:
   4
 Compartment 1:
 Sprayed
   3
  7.8000E+05
   1
   0
   0
   0
   0
 Compartment 2:
 Unsprayed
   3
  2.2000E+05
   0
   0
   1
   0
   0
 Compartment 3:
 Environ
   2
  0.0000E+00
   0
   0
   0
   0
   0
 Compartment 4:
 Control Rm
   1
  3.6210E+04
   0
   0
   1
   0
   0
 Pathways:
   6
 Pathway 1:
 Sprayed to Unsprayed
   1
   2
   2
 Pathway 2:
 Unsprayed to Sprayed
   2
   1
```

```
  2
Pathway 3:
Sprayed to Environ
  1
  3
  4
Pathway 4:
Unsprayed to Environ
  2
  3
  4
Pathway 5:
Environ to Control Rm
  3
  4
  2
Pathway 6:
Control Rm to Environ
  4
  3
  2
End of Plant Model File
Scenario Description Name:

Plant Model Filename:

Source Term:
  2
  1   7.8000E-01
  2   2.2000E-01
c:\data\shane\work\ginna\rhr flyup issue sept 2010\radtrad\ginna.inp
c:\data\shane\work\ginna\rhr flyup issue sept 2010\radtrad\loca.rft
  0.0000E+00
  0
  9.5000E-01   4.8500E-02   1.5000E-03   1.0000E+00
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
  4
Compartment 1:
  1
  1
  1
  0.0000E+00
  3
  0.0000E+00   0.0000E+00
  2.2000E-02   3.5000E+00
  8.7000E-01   0.0000E+00
  1
  0.0000E+00
  3
  0.0000E+00   0.0000E+00
  2.2000E-02   2.0000E+01
  8.7000E-01   0.0000E+00
  1
  0.0000E+00
  0
  0
```

```
  0
  0
  0
Compartment 2:
  1
  1
  0
  0
  0
  0
  1
  1.2000E+04
  3
  0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  1.3900E-02   9.5000E+01   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
Compartment 3:
  1
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 4:
  1
  1
  0
  0
  0
  0
  1
  5.4000E+03
  4
  0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  1.9400E-02   9.9000E+01   9.4000E+01   9.4000E+01
  2.0000E+00   9.9000E+01   9.4000E+01   9.4000E+01
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
Pathways:
  6
Pathway 1:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  1.3900E-02   4.8000E+04   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   4.8000E+04   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
```

```
Pathway 2:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  1.3900E-02   4.8000E+04   9.5000E+01   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   2.0000E-01
  2.4000E+01   1.0000E-01
  7.2000E+02   0.0000E+00
  0
Pathway 4:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   2.0000E-01
  2.4000E+01   1.0000E-01
  7.2000E+02   0.0000E+00
  0
Pathway 5:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.2000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  1.6700E-02   2.5000E+02   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   2.5000E+02   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
```

```
  0
  0
  0
  0
  0
  0
Pathway 6:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.2000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  1.6700E-02   2.5000E+02   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   2.5000E+02   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
EAB
  3
  1
  3
  0.0000E+00   2.1700E-04
  2.0000E+00   2.1700E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   3.4700E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  7.2000E+02   0.0000E+00
  0
Location 2:
LPZ
  3
  1
  6
  0.0000E+00   2.5100E-05
  2.0000E+00   2.5100E-05
  8.0000E+00   1.7800E-05
  2.4000E+01   8.5000E-06
  9.6000E+01   2.9300E-06
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   3.4700E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  7.2000E+02   0.0000E+00
  0
Location 3:
Control Rm
  4
  0
```

```
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  6
  0.0000E+00   1.7700E-03
  2.0000E+00   1.2500E-03
  8.0000E+00   4.8000E-04
  2.4000E+01   4.2400E-04
  9.6000E+01   3.6600E-04
  7.2000E+02   0.0000E+00
Simulation Parameters:
  3
  0.0000E+00   1.0000E-02
  2.0000E+00   1.0000E-01
  7.2000E+02   0.0000E+00
Output Filename:
C:\Data\Shane\Work\Ginna\2011 EAL Calc Revisions\CALC-2011-XXXX\Scoping\contleak.o0
  1
  1
  1
  0
  0
End of Scenario File
```

**LOCA.RFT**

```
Release Fraction and Timing Name:LOCA RFT
 Ginna, DA-NS-2001-087, R4, page 22 and RG 1.183, Table 4
 Duration (h):  Design Basis Accident
  0.0083      0.5000E+00  0.1300E+01  0.0000E+00  0.0000E+00
 Noble Gases:
  0.0         0.5000E-01  0.9500E+00  0.0000E+00  0.0000E+00
 Iodine:
  0.0         0.5000E-01  0.3500E+00  0.0000E+00  0.0000E+00
 Cesium:
  0.0         0.5000E-01  0.2500E+00  0.0000E+00  0.0000E+00
 Tellurium:
  0.0         0.0000E+00  0.5000E-01  0.0000E+00  0.0000E+00
 Strontium:
  0.0         0.0000E+00  0.2000E-01  0.0000E+00  0.0000E+00
 Barium:
  0.0         0.0000E+00  0.2000E-01  0.0000E+00  0.0000E+00
 Ruthenium:
  0.0         0.0000E+00  0.2500E-02  0.0000E+00  0.0000E+00
 Cerium:
  0.0         0.0000E+00  0.5000E-03  0.0000E+00  0.0000E+00
 Lanthanum:
  0.0         0.0000E+00  0.2000E-03  0.0000E+00  0.0000E+00
 Non-RadioactiAerosols (kg):
  0.0         0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 End of Release File
```

**Ginna LOCA63.nif**

```
Nuclide Inventory Name: Ginna LOCA
 Calc DA-NS-2002-037, R1
 Power Level:
  1.0
 Nuclides:
  63
Nuclide 001:
 Co-60
   7
  0.1663401096E+09
  0.6000E+02
  1.980E+02
 none     0.0000E+00
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 002:
 Kr-85m
   1
  0.1612800000E+05
  0.8500E+02
  7.51E+03
 Kr-85    0.2100E+00
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 003:
 Kr-85
   1
  0.3382974720E+09
  0.8500E+02
  3.23E+02
 none     0.0000E+00
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 004:
 Kr-87
   1
  0.4578000000E+04
  0.8700E+02
  1.447E+04
 Rb-87    0.1000E+01
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 005:
 Kr-88
   1
  0.1022400000E+05
  0.8800E+02
  2.032E+04
 Rb-88    0.1000E+01
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 006:
 Xe-131m
   1
  1.028E+06
  0.1310E+03
  3.087E+02
 none     0.0000E+00
 none     0.0000E+00
 none     0.0000E+00
```

```
Nuclide 007:
Xe-133m
  1
 1.89E+05
 0.1330E+03
 1.750E+03
Xe-133   1.0
none     0.0000E+00
none     0.0000E+00
Nuclide 008:
Xe-133
  1
 0.4531680000E+06
 0.1330E+03
 5.577E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 009:
Xe-135m
  1
 917.4
 0.1350E+03
 1.126E+04
Xe-135   0.9999E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 010:
Xe-135
  1
 0.3272400000E+05
 0.1350E+03
 1.414E+04
Cs-135   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 011:
Xe-138:
  1
 8.502E+2
 0.138E+03
 4.754E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 012:
I-131
  2
 0.6946560000E+06
 0.1310E+03
 2.805E+04
Xe-131m  0.1100E-01
none     0.0000E+00
none     0.0000E+00
Nuclide 013:
I-132
  2
 0.8280000000E+04
 0.1320E+03
 4.147E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
```

```
Nuclide 014:
I-133
  2
 0.7488000000E+05
 0.1330E+03
 5.687E+04
Xe-133m  0.2900E-01
Xe-133   0.9700E+00
none     0.0000E+00
Nuclide 015:
I-134
  2
 0.3156000000E+04
 0.1340E+03
 6.295E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 016:
I-135
  2
 0.2379600000E+05
 0.1350E+03
 5.367E+04
Xe-135m  0.1500E+00
Xe-135   0.8500E+00
none     0.0000E+00
Nuclide 017:
Cs-134
  3
 0.6507177120E+08
 0.1340E+03
 6.074E+03
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 018:
Cs-136
  3
 0.1131840000E+07
 0.1360E+03
 1.789E+03
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 019:
Cs-137
  3
 0.9467280000E+09
 0.1370E+03
 3.479E+03
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 020:
Rb-86
  3
 0.1612224000E+07
 0.8600E+02
 7.178E+01
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
```

```
Nuclide 021:
Te-127m
  4
 0.9417600000E+07
 0.1270E+03
 3.876E+02
Te-127   0.9800E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 022:
Te-127
  4
 0.3366000000E+05
 0.1270E+03
 2.982E+03
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 023:
Te-129m
  4
 0.2903040000E+07
 0.1290E+03
 1.314E+03
Te-129   0.6500E+00
I-129    0.3500E+00
none     0.0000E+00
Nuclide 024:
Te-129
  4
 0.4176000000E+04
 0.1290E+03
 8.945E+03
I-129    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 025:
Te-131m
  4
 0.1080000000E+06
 0.1310E+03
 4.042E+03
Te-131   0.2200E+00
I-131    0.7800E+00
none     0.0000E+00
Nuclide 026:
Te-132
  4
 0.2815200000E+06
 0.1320E+03
 3.987E+04
I-132    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 027:
Sb-127
  4
 0.3326400000E+06
 0.1270E+03
 3.009E+03
Te-127m  0.1800E+00
Te-127   0.8200E+00
none     0.0000E+00
```

```
Nuclide 028:
Sb-129
  4
 0.1555200000E+05
 0.1290E+03
 9.001E+03
Te-129m  0.2200E+00
Te-129   0.7700E+00
none     0.0000E+00
Nuclide 029:
Sr-89
  5
 0.4363200000E+07
 0.8900E+02
 2.733E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 030:
Sr-90
  5
 0.9189573120E+09
 0.9000E+02
 2.557E+03
Y-90     0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 031:
Sr-91
  5
 0.3420000000E+05
 0.9100E+02
 3.424E+04
Y-91m    0.5800E+00
Y-91     0.4200E+00
none     0.0000E+00
Nuclide 032:
Sr-92
  5
 0.9756000000E+04
 0.9200E+02
 3.700E+04
Y-92     0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 033:
Ba-139
  6
 0.4962000000E+04
 0.1390E+03
 5.141E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 034:
Ba-140
  6
 0.1100736000E+07
 0.1400E+03
 4.936E+04
La-140   0.1000E+01
none     0.0000E+00
none     0.0000E+00
```

```
Nuclide 035:
Ru-103
  7
 0.3393792000E+07
 0.1030E+03
 4.257E+04
Rh-103m  0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 036:
Ru-105
  7
 0.1598400000E+05
 0.1050E+03
 2.888E+04
Rh-105   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 037:
Ru-106
  7
 0.3181248000E+08
 0.1060E+03
 1.441E+04
Rh-106   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 038:
Rh-105
  7
 0.1272960000E+06
 0.1050E+03
 2.584E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 039:
Mo-99
  7
 0.2376000000E+06
 0.9900E+02
 5.340E+04
Tc-99m   0.8800E+00
Tc-99    0.1200E+00
none     0.0000E+00
Nuclide 040:
Tc-99m
  7
 0.2167200000E+05
 0.9900E+02
 4.688E+04
Tc-99    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 041:
Ce-141
  8
 0.2808086400E+07
 0.1410E+03
 4.682E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
```

```
Nuclide 042:
Ce-143
  8
 0.1188000000E+06
 0.1430E+03
 4.357E+04
Pr-143    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 043:
Ce-144
  8
 0.2456352000E+08
 0.1440E+03
 3.567E+04
Pr-144m   0.1800E-01
Pr-144    0.9800E+00
none      0.0000E+00
Nuclide 044:
Pu-238
  8
 0.2768863824E+10
 0.2380E+03
 1.231E+02
U-234     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 045:
Pu-239
  8
 0.7594336440E+12
 0.2390E+03
 1.022E+01
U-235     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 046:
Pu-240
  8
 0.2062920312E+12
 0.2400E+03
 1.541E+01
U-236     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 047:
Pu-241
  8
 0.4544294400E+09
 0.2410E+03
 3.396E+03
U-237     0.2400E-04
Am-241    0.1000E+01
none      0.0000E+00
Nuclide 048:
Np-239
  8
 0.2034720000E+06
 0.2390E+03
 5.798E+05
Pu-239    0.1000E+01
none      0.0000E+00
none      0.0000E+00
```

```
Nuclide 049:
Y-90
  9
 0.2304000000E+06
 0.9000E+02
 2.673E+03
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 050:
Y-91
  9
 0.5055264000E+07
 0.9100E+02
 3.523E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 051:
Y-92
  9
 0.1274400000E+05
 0.9200E+02
 3.716E+4
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 052:
Y-93
  9
 0.3636000000E+05
 0.9300E+02
 4.268E+04
Zr-93    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 053:
Nb-95
  9
 0.3036960000E+07
 0.9500E+02
 4.782E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 054:
Zr-95
  9
 0.5527872000E+07
 0.9500E+02
 4.749E+04
Nb-95m   0.780E-02
Nb-95    0.992E+00
none     0.0000E+00
Nuclide 055:
Zr-97
  9
 0.6084000000E+05
 0.9700E+02
 4.743E+04
Nb-97m   0.9470E+00
Nb-97    0.5300E-01
none     0.0000E+00
```

```
Nuclide 056:
La-140
  9
 0.1449792000E+06
 0.1400E+03
 5.113E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 057:
La-141
  9
 0.1414800000E+05
 0.1410E+03
 4.611E+04
Ce-141   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 058:
La-142
  9
 0.5550000000E+04
 0.1420E+03
 4.539E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 059:
Nd-147
  9
 0.9486720000E+06
 0.1470E+03
 1.872E+04
Pm-147   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 060:
Pr-143
  9
 0.1171584000E+07
 0.1430E+03
 4.274E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 061:
Am-241
  9
 0.1363919472E+11
 0.2410E+03
 4.169E+00
Np-237   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 062:
Cm-242
  9
 0.1406592000E+08
 0.2420E+03
 9.663E+02
Pu-238   0.1000E+01
none     0.0000E+00
none     0.0000E+00
```

```
Nuclide 063:
Cm-244
  9
 0.571508136E+9
 0.2440E+03
 1.137E+02
Pu-240    0.1000E+01
none      0.0000E+00
none      0.0000E+00
End of Nuclear Inventory File
```

## Ginna.inp

```
Dose Conversion Factors from FGRs 11 & 12
Worst Lung Clearance Class for CEDE - nuclides consistent with DA-NS-2002-037, R1 Co-
58 and 60 are omitted
         9 ORGANS DEFINED IN THIS FILE:
GONADS
BREAST
LUNGS
RED MARR
BONE SUR
THYROID
REMAINDER
EFFECTIVE
SKIN(FGR)
        63 NUCLIDES DEFINED IN THIS FILE:
Co-60
Kr-85m
Kr-85
Kr-87
Kr-88
Xe-131m
Xe-133m
Xe-133
Xe-135m
Xe-135
Xe-138
I-131
I-132
I-133
I-134
I-135
Cs-134
Cs-136
Cs-137    Actually Ba-137m for EDE
Rb-86
Te-127m
Te-127
Te-129m
Te-129
Te-131m
Te-132
Sb-127
Sb-129
Sr-89
Sr-90
Sr-91
Sr-92
Ba-139
```

Ba-140
Ru-103
Ru-105
Ru-106
Rh-105
Mo-99
Tc-99m
Ce-141
Ce-143
Ce-144
Pu-238
Pu-239
Pu-240
Pu-241
Np-239
Y-90
Y-91
Y-92
Y-93
Nb-95
Zr-95
Zr-97
La-140
La-141 added
La-142
Nd-147
Pr-143
Am-241
Cm-242
Cm-244

| | CLOUDSHINE | GROUND SHINE 8HR | GROUND SHINE 7DAY | GROUND SHINE RATE | INHALED ACUTE | INHALED CHRONIC | INGESTION |
|---|---|---|---|---|---|---|---|
| Co-60 | | | | | | | |
| GONADS | 1.230E-13 | 7.056E-11 | 1.480E-09 | 2.450E-15 | -1.000E+00 | 4.760E-09 | 3.190E-09 |
| BREAST | 1.390E-13 | 6.739E-11 | 1.413E-09 | 2.340E-15 | -1.000E+00 | 1.840E-08 | 1.100E-09 |
| LUNGS | 1.240E-13 | 6.537E-11 | 1.371E-09 | 2.270E-15 | -1.000E+00 | 3.450E-07 | 8.770E-10 |
| RED MARR | 1.230E-13 | 6.710E-11 | 1.407E-09 | 2.330E-15 | -1.000E+00 | 1.720E-08 | 1.320E-09 |
| BONE SUR | 1.780E-13 | 8.956E-11 | 1.879E-09 | 3.110E-15 | -1.000E+00 | 1.350E-08 | 9.390E-10 |
| THYROID | 1.270E-13 | 6.480E-11 | 1.359E-09 | 2.250E-15 | -1.000E+00 | 1.620E-08 | 7.880E-10 |
| REMAINDER | 1.200E-13 | 6.508E-11 | 1.365E-09 | 2.260E-15 | -1.000E+00 | 3.600E-08 | 4.970E-09 |
| EFFECTIVE | 1.260E-13 | 6.768E-11 | 1.419E-09 | 2.350E-15 | -1.000E+00 | 5.910E-08 | 2.770E-09 |
| SKIN(FGR) | 1.450E-13 | 7.948E-11 | 1.667E-09 | 2.760E-15 | -1.000E+00 | 0.000E+00 | 0.000E+00 |
| Kr-85m | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.480E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Kr-85 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.190E-16 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Kr-87 | | | | | | | |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 4.120E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Kr-88 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.020E-13 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Xe-131m | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 3.890E-16 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Xe-133m | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.370E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Xe-133 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.560E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Xe-135m | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.040E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Xe-135 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.190E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Xe-138 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 5.770E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| I-131 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.820E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 8.890E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| I-132 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.120E-13 | 0.0 | 0.0 | 0.0 | 0.0 | 1.030E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| I-133 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.940E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 1.580E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| I-134 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.300E-13 | 0.0 | 0.0 | 0.0 | 0.0 | 3.550E-11 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| I-135 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.980E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 3.320E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Cs-134 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.570E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 1.250E-08 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Cs-136 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.060E-13 | 0.0 | 0.0 | 0.0 | 0.0 | 1.980E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Cs-137 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.880E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 8.630E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Rb-86 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 4.810E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 1.790E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Te-127m | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.470E-16 | 0.0 | 0.0 | 0.0 | 0.0 | 5.810E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Te-127 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.420E-16 | 0.0 | 0.0 | 0.0 | 0.0 | 8.600E-11 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| Te-129m | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.550E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 6.470E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Te-129 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.750E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 2.420E-11 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Te-131m | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.010E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 1.730E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Te-132 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.030E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 2.550E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Sb-127 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 3.330E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 1.630E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Sb-129 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.140E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 1.740E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Sr-89 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.730E-17 | 0.0 | 0.0 | 0.0 | 0.0 | 1.120E-08 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Sr-90 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.530E-18 | 0.0 | 0.0 | 0.0 | 0.0 | 3.510E-07 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Sr-91 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 3.450E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 4.490E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Sr-92 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 6.790E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 2.180E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Ba-139 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.170E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 4.640E-11 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Ba-140 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 8.580E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 1.010E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Ru-103 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.250E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 2.420E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Ru-105 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 3.810E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 1.230E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Ru-106 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 1.290E-07 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Rh-105 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 3.720E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 2.580E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Mo-99 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.280E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 1.070E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Tc-99m | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 5.890E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 8.800E-12 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Ce-141 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 3.430E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 2.420E-09 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Ce-143 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.290E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 9.160E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Ce-144 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 8.530E-16 | 0.0 | 0.0 | 0.0 | 0.0 | 1.010E-07 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Pu-238 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 4.880E-18 | 0.0 | 0.0 | 0.0 | 0.0 | 1.060E-04 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Pu-239 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 4.240E-18 | 0.0 | 0.0 | 0.0 | 0.0 | 1.160E-04 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Pu-240 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 4.750E-18 | 0.0 | 0.0 | 0.0 | 0.0 | 1.160E-04 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Pu-241 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.250E-20 | 0.0 | 0.0 | 0.0 | 0.0 | 2.230E-06 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Np-239 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 7.690E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 6.780E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Y-90 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.900E-16 | 0.0 | 0.0 | 0.0 | 0.0 | 2.280E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Y-91 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.600E-16 | 0.0 | 0.0 | 0.0 | 0.0 | 1.320E-08 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Y-92 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.300E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 2.110E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Y-93 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 4.800E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 5.820E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Nb-95 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 3.740E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 1.570E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Zr-95 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 3.600E-14 | 0.0 | 0.0 | 0.0 | 0.0 | 6.390E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Zr-97 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 9.020E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 1.170E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| La-140 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.170E-13 | 0.0 | 0.0 | 0.0 | 0.0 | 1.310E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| La-141 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 2.390E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 1.57E-10 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| La-142 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 1.440E-13 | 0.0 | 0.0 | 0.0 | 0.0 | 6.840E-11 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Nd-147 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 6.190E-15 | 0.0 | 0.0 | 0.0 | 0.0 | 1.850E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Pr-143 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| EFFECTIVE | 2.100E-17 | 0.0 | 0.0 | 0.0 | 0.0 | 2.190E-09 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Am-241 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 8.180E-16 | 0.0 | 0.0 | 0.0 | 0.0 | 1.200E-04 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Cm-242 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 5.690E-18 | 0.0 | 0.0 | 0.0 | 0.0 | 4.670E-06 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Cm-244 | | | | | | | |
| GONADS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BREAST | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| LUNGS | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| RED MARR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| BONE SUR | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| THYROID | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| REMAINDER | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| EFFECTIVE | 4.910E-18 | 0.0 | 0.0 | 0.0 | 0.0 | 6.700E-05 | 0.0 |
| SKIN(FGR) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

**ATTACHMENT 1, DESIGN VERIFICATION REPORT (FOR INFORMATION ONLY)**

| **Doc Type being verified:** | ☐ Design Change ☒ Calculation ☐ Specification<br>☐ Other: | | |
|---|---|---|---|
| **Document No.:** | **CALC-2011-0020** | **Rev.:** | 000 |

**Extent of Design Verification** (Briefly describe):

Checked inputs, spreadsheet, and RADTRAD input and output.

**Method of Design Verification:**

☒ Design Review ☐ Qualification Testing

☐ Alternate Calculations ☐ Applicability of Proven Design

**Results of Design Verification:**

☐ Fully acceptable with no issues identified

☒ Fully acceptable based on the following issues identified and resolved:

| No. | Verifier's Comment | Preparer's Response (Indicate if not required) | Verifier Concurs? (Yes/No) |
|---|---|---|---|
| 1 | Table 4-1 – Plant vent flow rate is cited as 77,000 cfm per EPIP-2-4. EPIP-2-4 indicates plant vent flow rate of 77,844 cfm normal and 71,289 cfm emergency. EPIP-2-3 cites 76,000 cfm. Use of slightly higher flow rate produces no change in EAL within the number of significant digits specified. Use of the emergency flow rate results in a slightly less conservative (higher) EAL. | This value is left unchanged. It is made an assumption. The reason is because there are several values for the actual flow rate. DA-RP-2001-017 states the design flow rate is 75,000 cfm, however flow test acceptance criterion is greater than 80,000 cfm. Apparently the EPIPs use best-estimate values. Rather than try to use a single value from reference, the value of 77,000 cfm is used to encompass the variability. | Y |
| 2 | Table 4-1 and Section 5 - Plant vent X/Q is assumed to apply to ARV and MSSVs, but no discussion is provided in the assumption section. | Added Assumption 5.6. | Y |
| 3 | Table 4-1 – Section 5.13 of EPIP-2-3 rev 18 indicates 2565 cfm (153,900 cfh) for ARV flow rate rather than 2818 cfm. Also, 5550 cfm (333,000 cfh) flow rate per safety valve (4 available) rather than 5615 cfm. | The values used were based on the EPIP-2-3 value in cc/sec divided by the conversion factor of 471.947. There appears to be a discrepancy in the conversion in EPIP-2-3. Values and calculation have been updated. | Y |
| 4 | Table 4-1 – P-9 shows limit for R-12 with 2 fans is 2.68E6 cpm rather than 2.56E6 cpm. | The cited reference is DA-RP-98-091. The calculation is below =500/(294*0.0000016*0 .000000055*7550000) on page 11. This may be an error in P-9. Since the calculation is the basis for the ODCM setpoint no change is made. | Y |
| 5 | Reference 6.1 – provide a revision number (9807). | Corrected. | Y |

| | | | |
|---|---|---|---|
| Lead Design Verifier: | J. R. MASSARI | [signature] | 12/19/2011 |
| | Name | Signature | Date |
| Engineering Manager (if required): N/A | | | |
| Discipline Design Verifiers if required: N/A | | | |
| | | | |
| | | | |
| | | | |
| Discipline | Name | Signature | Date |

(USE ADDITIONAL SHEETS AS REQUIRED)

Check if Design Review Checklist is attached ☒

Check if additional sheets are used ☐

## ATTACHMENT 2, DESIGN VERIFICATION CHECKLIST (FOR INFORMATION ONLY)

*The following questions are required to be addressed based on Constellation Nuclear Generation commitment to* ANSI/ASME NQA-1-1994 for design verification activities. This checklist is intended to assist when using the Design Review method of design verification to ensure relevant items are addressed in the verification effort. Each "No" answer will require correction or resolution by the originator of the document being verified prior to full acceptance by the design verifier(s).

Doc #: **CALC-2011-0020** Rev 000 Lead Design Verifier's Name: J. R. MASSARI

| | Review Check | | |
|---|---|---|---|
| | Yes | No | N/A |
| 1. Were the design inputs correctly selected? | X | | |
| 2. Are assumptions necessary to perform the design activity adequately described and reasonable? | X | | |
| Where necessary, are the assumptions identified for subsequent re-verifications when the detailed design activities are completed? | | | X |
| 3. Was an appropriate design method used? | X | | |
| 4. Were the design inputs correctly incorporated into the design? | X | | |
| 5. Is the design output reasonable compared to design inputs? | X | | |
| 6. Are the necessary design input and verification requirements for interfacing organizations specified in the design documents or in supporting procedures or instructions? | | | X |

ATTACHMENT (5)

EAL WALLCHART

# THIS PAGE IS AN OVERSIZED DRAWING OR FIGURE,

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**"GINNA STATION EVENT EVALUATION & CLASSIFICATION REV XX"**
**PG 1 OF 2**

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**D-01**

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**D-02**