<u>Attachment 2</u>

EP-AA-1012, Addendum 3, Revision 5, *"R. E. Ginna Nuclear Power Plant Emergency Action Levels"*

Emergency Plan Addendum Revision

R.E. Ginna Nuclear Power Plant
Emergency Action Levels

GINNA STATION EVENT EVALUATION AND CLASSIFICATION

1.0 PURPOSE

1.1 The purpose of this procedure is to provide guidance to personnel in evaluating situations which may require activation of the Nuclear Emergency Response Plan and direct them to appropriate implementing procedures. Prompt recognition and declaration is necessary to ensure the timely activation of support functions and notification of offsite organizations.

2.0 RESPONSIBILITY

2.1 This Shift Manager/Emergency Director (SM/ED) is responsible for initiating this procedure.

2.2 Once the EOF assumes command and control of the emergency, the ED becomes responsible for continuing this procedure.

3.0 REFERENCES

3.1 Developmental References

3.1.1 10CFR50 Appendix E

3.1.2 NUREG-0654, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

3.1.3 NUREG-0696, Functional Criteria for Emergency Response Facilities

3.1.4 Ginna Station Nuclear Emergency Response Plan

3.1.5 NEI 99-01 Revision 5, Methodology for Development of Emergency Action Levels

3.1.6 R.E. Ginna EAL Technical Basis

3.2 Implementing References

3.2.1 None

4.0 PRECAUTIONS

4.1 Once indications are available to Control Room operators that an EAL has been exceeded, the Shift Manager should classify the event within 15 minutes.

4.2 In the event that multiple "Initiating Conditions" are identified, the SM/ED shall review each condition and classify according to the highest Emergency Classification Level obtained.

4.3 During any event, the entire procedure should be reviewed for possible reclassification of the event.

4.4 Any time a current set of conditions is identified which meets the criteria for an Emergency Classification, the event shall be classified and declared, even if the condition identified is quickly corrected.

5.0 PREREQUISITES:

5.1 Entry to this procedure may be directed by various other plant procedures or at the discretion of the SM/ED.

6.0 ACTIONS:

6.1 In the event of an abnormal condition the Control Room Personnel will:

6.1.1 Perform the immediate responses defined in the appropriate plant procedures.

6.1.2 Identify the <u>initiating conditions</u> using either the guidelines of the EAL wall chart or Attachment 1 of this procedure.

6.1.3 Implement applicable Emergency Plan classification procedures.

6.2 Periodically re-evaluate the condition after initial classification of the event using the EAL wall chart or Attachment 1.

6.3 At the conclusion of the event, refer to emergency termination and recovery procedures.

6.4 Any time previous initiating conditions are identified that would have warranted an Emergency Classification but they are no longer in effect at the time of identification, and do not require further evaluation or analysis, the event will be classified, but not declared.

6.4.1 Conditions which are corrected, but may require further safety evaluation or analysis, will be classified and declared.

6.4.2 Conditions which depend on delayed evaluation results, e.g., chemistry, RP analysis, etc., shall be classified and declared as soon as the results are known.

6.4.3 The NRC will be notified any time an event is classified.

6.4.4 The Director, Emergency Preparedness (or alternate) shall be informed of the NRC notification by the Control Room, as soon possible to provide "courtesy notifications" to Wayne County, Monroe County and New York State. For events that have been classified but not declared, there is no 15 minute requirement for notifications.

7.0 ATTACHMENTS

1. Emergency Action Levels (EALs)
2. EAL Wall Chart (sample)
3. Technical Basis

ATTACHMENT 1

EMERGENCY ACTION LEVELS

EMERGENCY ACTION LEVELS (EALs)

INDEX

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

R – Abnormal Rad Release/Rad Effluent

1 – Offsite Rad Conditions

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode
1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D
RG1.1 **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)	**RS1.1** **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)	**RA1.1** **ANY** gaseous monitor reading > Table R-1 column "Alert" for ≥ 15 min. (Note 2)	**RU1.1** **ANY** gaseous or liquid monitor reading > Table R-1 column "UE" for ≥ 60 min. (Note 2)

R – Abnormal Rad Release/Rad Effluent

1 – Offsite Rad Conditions

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D
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	**RS1.2** Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary	**RA1.2** Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x P-9 limits for ≥ 15 min. (Note 2)	**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)

R – Abnormal Rad Release/Rad Effluent

1 – Offsite Rad Conditions

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode
1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D
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 (Note 1) **OR** Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr 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 (Note 1) **OR** Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the site boundary	NONE	NONE

R – Abnormal Rad Release/Rad Effluent

2 – Onsite Rad Conditions & Spent Fuel Events

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D
NONE	NONE	**RA2.1** 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	**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) **AND** Area radiation monitor reading rise on **EITHER**: R-2 Containment **OR** R-5 Spent Fuel Pool

R – Abnormal Rad Release/Rad Effluent

2 – Onsite Rad Conditions & Spent Fuel Events

<table>
<tr><th colspan="7">GENERAL EMERGENCY</th><th colspan="7">SITE AREA EMERGENCY</th><th colspan="7">ALERT</th><th colspan="7">UNUSUAL EVENT</th></tr>
<tr><td colspan="7">Mode</td><td colspan="7">Mode</td><td colspan="7">Mode</td><td colspan="7">Mode</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>D</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>D</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>D</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>D</td></tr>
<tr><td colspan="7">NONE</td><td colspan="7">NONE</td><td colspan="7">RA2.2
A water level drop in a reactor refueling pathway that will result in irradiated fuel becoming uncovered</td><td colspan="7">RU2.2
Unplanned area radiation reading increases by a factor of 1,000 over normal levels</td></tr>
</table>

R – Abnormal Rad Release/Rad Effluent

3 – CR/CAS Rad

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode
1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D
NONE	NONE	**RA3.1** 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	NONE

H – Hazards and Other Conditions Affecting Plant Safety

1 – Natural or Destructive Phenomena

GENERAL EMERGENCY							SITE AREA EMERGENCY							ALERT							UNUSUAL EVENT						
Mode							Mode							Mode							Mode						
1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D
NONE							NONE							**HA1.1** Earthquake confirmed by **EITHER**: Earthquake felt in plant **OR** National Earthquake Information Center (Note 6) **AND** Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area							**HU1.1** Seismic event identified by **ANY two** of the following: • Event indicator on seismograph indicates seismic event detected • Earthquake felt in plant • National Earthquake Information Center (Note 6)						

H – Hazards and Other Conditions Affecting Plant Safety

1 – Natural or Destructive Phenomena

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 1 2 3 4 5 6 D	Mode: 1 2 3 4 5 6 D	Mode: 1 2 3 4 5 6 D	Mode: 1 2 3 4 5 6 D
NONE	NONE	**HA1.2** 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	**HU1.2** Tornado striking within Protected Area boundary **OR** Sustained high winds > 75 mph

H – Hazards and Other Conditions Affecting Plant Safety

1 – Natural or Destructive Phenomena

<table>
<tr><th colspan="7">GENERAL EMERGENCY</th><th colspan="7">SITE AREA EMERGENCY</th><th colspan="7">ALERT</th><th colspan="7">UNUSUAL EVENT</th></tr>
<tr><th colspan="7">Mode</th><th colspan="7">Mode</th><th colspan="7">Mode</th><th colspan="7">Mode</th></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>D</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>D</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>D</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>D</td></tr>
<tr><td colspan="7">NONE</td><td colspan="7">NONE</td><td colspan="7">HA1.3
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</td><td colspan="7">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</td></tr>
</table>

H – Hazards and Other Conditions Affecting Plant Safety

1 – Natural or Destructive Phenomena

GENERAL EMERGENCY							SITE AREA EMERGENCY							ALERT							UNUSUAL EVENT						
Mode							Mode							Mode							Mode						
1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D
NONE							NONE							**HA1.4** 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							**HU1.4** Turbine failure resulting in casing penetration or damage to turbine or generator seals						

H – Hazards and Other Conditions Affecting Plant Safety

1 – Natural or Destructive Phenomena

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode
1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D
NONE	NONE	**HA1.5** Lake level > 253 ft **OR** Screen House Suction Bay water level < 16 ft or < 14.5 ft by manual level measurement	**HU1.5** 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

H – Hazards and Other Conditions Affecting Plant Safety

1 – Natural or Destructive Phenomena

GENERAL EMERGENCY							SITE AREA EMERGENCY							ALERT							UNUSUAL EVENT						
Mode							Mode							Mode							Mode						
1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D
NONE							NONE							**HA1.6** 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							NONE						

H – Hazards and Other Conditions Affecting Plant Safety

2 – Fire or Explosion

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode
1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D	1 | 2 | 3 | 4 | 5 | 6 | D
NONE	NONE	**HA2.1** 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	**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)

H – Hazards and Other Conditions Affecting Plant Safety

2 – Fire or Explosion

GENERAL EMERGENCY							SITE AREA EMERGENCY							ALERT							UNUSUAL EVENT						
Mode							Mode							Mode							Mode						
1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D
NONE							NONE							NONE							**HU2.2** Explosion within the Protected Area						

H – Hazards and Other Conditions Affecting Plant Safety

3 – Hazardous Gas

GENERAL EMERGENCY							SITE AREA EMERGENCY							ALERT							UNUSUAL EVENT						
Mode							Mode							Mode							Mode						
1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D
NONE							NONE							**HA3.1 Alert** Access to ANY of the following areas is impeded due to toxic, corrosive, asphyxiant or flammable gases (Note 5): - Control Room (Note 8) - Auxiliary Building							**HU3.1** Release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations						

H – Hazards and Other Conditions Affecting Plant Safety

3 – Hazardous Gas

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 1 2 3 4 5 6 D	Mode: 1 2 3 4 5 6 D	Mode: 1 2 3 4 5 6 D	Mode: 1 2 3 4 5 6 D
NONE	NONE	NONE	**HU3.2** Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event

H – Hazards and Other Conditions Affecting Plant Safety

4 – Security

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D	Mode 1 | 2 | 3 | 4 | 5 | 6 | D
HG4.1 A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions	**HS4.1** A hostile action is occurring or has occurred within the Protected Area as reported by Security Shift Supervision	**HA4.1** 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	**HU4.1** 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

H – Hazards and Other Conditions Affecting Plant Safety

4 – Security

GENERAL EMERGENCY							SITE AREA EMERGENCY							ALERT							UNUSUAL EVENT						
Mode							Mode							Mode							Mode						
1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D	1	2	3	4	5	6	D
HG4.2 A hostile action has caused failure of Spent Fuel Cooling systems **AND** Imminent fuel damage is likely							NONE							NONE							NONE						

H – Hazards and Other Conditions Affecting Plant Safety

5 – Control Room Evacuation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 1 | 2 | 3 | 4 | 5 | 6 | D	Mode: 1 | 2 | 3 | 4 | 5 | 6 | D	Mode: 1 | 2 | 3 | 4 | 5 | 6 | D	Mode: 1 | 2 | 3 | 4 | 5 | 6 | D
NONE	**HS5.1** Control Room evacuation has been initiated **AND** Control of the plant **cannot** be established within 35 min.	**HA5.1** Control Room evacuation has been initiated	NONE

H – Hazards and Other Conditions Affecting Plant Safety

6 – Judgment

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 1 | 2 | 3 | 4 | 5 | 6 | D	Mode: 1 | 2 | 3 | 4 | 5 | 6 | D	Mode: 1 | 2 | 3 | 4 | 5 | 6 | D	Mode: 1 | 2 | 3 | 4 | 5 | 6 | D
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	**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	**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)	**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

E – ISFSI

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode N/A
NONE	NONE	NONE	**EU1.1** Damage to a loaded cask confinement boundary

S – System Malfunction

1 – Loss of AC Power

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode
1 | 2 | 3 | 4	1 | 2 | 3 | 4	1 | 2 | 3 | 4	1 | 2 | 3 | 4
SG1.1 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 (Note 4) **OR** **ORANGE** or **RED** path condition exists F-0.2 Core Cooling	**SS1.1** Loss of **all** offsite and **all** onsite AC power, Table S-1 to 480V safeguards buses for ≥ 15 min. (Note 4)	**SA1.1** 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	**SU1.1** Loss of **all** offsite AC power, Table S-1, to 480V safeguards buses for ≥ 15 min. (Note 4)

S – System Malfunction

2 – Loss of DC Power

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode 1 2 3 4	Mode 1 2 3 4	Mode 1 2 3 4	Mode 1 2 3 4
NONE	**SS2.1** < 108 VDC on **both** 125 VDC buses 1A and 1B for ≥ 15 min. (Note 4)	NONE	NONE

S – System Malfunction

3 – Criticality & RPS Failure

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 1	Mode: 1	Mode: 1	Mode: 3, 4
SG3.1 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	**SS3.1** 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%	**SA3.1** An automatic trip failed to shut down the reactor as indicated by reactor power > 5% **AND** Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power ≤ 5%	**SU3.1** An unplanned sustained positive startup rate observed on nuclear instrumentation

S – System Malfunction

4 – Inability to Reach or Maintain Shutdown Conditions

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode 1 2 3 4	Mode 1 2 3 4	Mode 1 2 3 4	Mode 1 2 3 4
NONE	NONE	NONE	**SU4.1** Plant is **not** brought to required operating mode within Technical Specifications LCO required action completion time

S – System Malfunction

5 – Instrumentation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode
1 2 3 4	1 2 3 4	1 2 3 4	1 2 3 4
NONE	**SS5.1** 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)	**SA5.1** 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)	**SU5.1** Unplanned loss of 6 or more annunciator panels, Table S-2, or >75% of MCB indications for ≥ 15 min. (Note 4)

S – System Malfunction

6 – Communications

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode 1 | 2 | 3 | 4	Mode 1 | 2 | 3 | 4	Mode 1 | 2 | 3 | 4	Mode 1 | 2 | 3 | 4
NONE	NONE	NONE	**SU6.1** 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

S – System Malfunction

7 – Fuel Clad Degradation

GENERAL EMERGENCY							SITE AREA EMERGENCY							ALERT							UNUSUAL EVENT						
Mode							Mode							Mode							Mode						
1	2	3	4				1	2	3	4				1	2	3	4				1	2	3	4			
NONE							NONE							NONE							**SU7.1** RCS specific activity > 60 µCi/gm dose equivalent I-131						

S – System Malfunction

7 – Fuel Clad Degradation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 1 2 3 4	Mode: 1 2 3 4	Mode: 1 2 3 4	Mode: 1 2 3 4
NONE	NONE	NONE	**SU7.2** Valid Letdown Monitor (R-9) reading ≥ 4.8 R/hr

S – System Malfunction

8 – RCS Leakage

GENERAL EMERGENCY							SITE AREA EMERGENCY							ALERT							UNUSUAL EVENT						
Mode							Mode							Mode							Mode						
1	2	3	4				1	2	3	4				1	2	3	4				1	2	3	4			
NONE							NONE							NONE							**SU8.1** Unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Notes 4 ,7) **OR** Identified leakage > 25 gpm for ≥ 15 min. (Notes 4, 7)						

F – Fission Product Barrier Degradation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 1 2 3 4	Mode: 1 2 3 4	Mode: 1 2 3 4	Mode: 1 2 3 4
FG1.1 Loss of **ANY** two barriers **AND** Loss or potential loss of the third barrier (Table F-1)	**FS1.1** Loss or potential loss of **ANY** two barriers (Table F-1)	**FA1.1** **ANY** loss or **ANY** potential loss of either Fuel Clad or RCS (Table F-1)	**FU1.1** **ANY** loss or **ANY** potential loss of Containment (Table F-1)

C – Cold Shutdown/Refueling System Malfunction

1 – Loss of AC Power

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6, D
NONE	NONE	**CA1.1** Loss of **all** offsite and **all** onsite AC power, Table C-1, to 480V safeguards buses for ≥ 15 min. (Note 4)	**CU1.1** 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

C – Cold Shutdown/Refueling System Malfunction

2 – Loss of DC Power

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6
NONE	NONE	NONE	**CU2.1** < 108 VDC on **required** 125 VDC buses for ≥ 15 min. (Note 4)

C – Cold Shutdown/Refueling System Malfunction

3 – RCS Level

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 5, 6	Mode: 5, 6	Mode: 5, 6	Mode: 5
CG3.1 RCS level **cannot** be monitored for ≥ 30 min. with core uncovery indicated by **ANY** of the following (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	**CS3.1** RCS level **cannot** be monitored for ≥ 30 min. with core uncovery, as indicated by **ANY** of the following (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	**CA3.1** 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)	**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)

C – Cold Shutdown/Refueling System Malfunction

3 – RCS Level

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6, D	Mode: 6
NONE	NONE	NONE	**CU3.2** 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)

C – Cold Shutdown/Refueling System Malfunction

3 – RCS Level

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	Mode	Mode	Mode
5, 6, D	5, 6, D	5, 6, D	6
NONE	NONE	NONE	**CU3.3** 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

C – Cold Shutdown/Refueling System Malfunction

4 – RCS Temperature

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode	**Mode**	**Mode**	**Mode**
5 6 D	**5 6 D**	**5 6**	**5 6**
NONE	NONE	**CA4.1** 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)	**CU4.1** Unplanned event results in RCS temperature > 200°F

C – Cold Shutdown/Refueling System Malfunction

4 – RCS Temperature

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6
NONE	NONE	NONE	**CU4.2** Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 4)

C – Cold Shutdown/Refueling System Malfunction

5 – Communications

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6, D
NONE	NONE	NONE	**CU5.1** 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

C – Cold Shutdown/Refueling System Malfunction

6 – Inadvertent Criticality

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6, D	Mode: 5, 6
NONE	NONE	NONE	**CU6.1** An unplanned sustained positive startup rate observed on nuclear instrumentation

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	6.54E+6 cpm w/ 1 fan* 4.54E+6 cpm w/ 2 fans*
CNMT Vent Noble Gas Hi Range **(R-12A)**	1.40E+2 µC/cc	1.40E+1 µC/cc	1.40E+0 µC/cc	N/A
Plant Vent Noble Gas **(R-14)**	N/A	N/A	N/A	5.80E+5 cpm
Plant Vent Noble Gas Hi Range **(R-14A)**	1.27E+1 µC/cc	1.27E+0 µC/cc	1.27E-1 µC/cc	N/A
Air Ejector Noble Gas **(R-15)**	N/A	N/A	N/A	5.30E+5 cpm
Air Ejector Noble Gas Hi Range **(R-48)**	1.20E+2 µC/cc	1.20E+1 µC/cc	1.20E+0 µC/cc	N/A
Main Steam Line **(R-31/R-32)** 1 ARV 1 Safety 2 Safety 3 Safety 4 Safety	 1.2E+3 mR/hr 5.7E+2 mR/hr 2.8E+2 mR/hr 1.9E+2 mR/hr 1.4E+2 mR/hr	 1.2E+2 mR/hr 5.7E+1 mR/hr 2.8E+1 mR/hr 1.9E+1 mR/hr 1.4E+1 mR/h	 1.2E+1 mR/hr 5.7E+0 mR/hr 2.8E+0 mR/hr 1.9E+0 mR/hr 1.4E+0 mR/hr	 8.0E+0 mR/hr 3.7E+0 mR/hr N/A N/A 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)** **(R-20B)**	 N/A N/A	 N/A N/A	 N/A N/A	 4.0E+4 cpm 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

* For containment purge

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

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

<table>
<tr><th colspan="3">Table C-4 RCS Reheat Duration Thresholds</th></tr>
<tr><th>RCS Status</th><th>Containment Closure Status</th><th>Duration</th></tr>
<tr><td>Intact AND not reduced inventory</td><td>N/A</td><td>60 min.*</td></tr>
<tr><td rowspan="2">Not intact OR reduced inventory</td><td>Established</td><td>20 min.*</td></tr>
<tr><td>Not established</td><td>0 min.</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable</td></tr>
</table>

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

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

Table S-1 AC Power Sources	
Onsite	• EDG 1A (Bus 14) • EDG 1B (Bus 16)
Offsite	• Station Auxiliary Transformer 12A • Station Auxiliary Transformer 12B • Unit Auxiliary Transformer 11 backfeed (if currently established)

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

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

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 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 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. 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) 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 2. Containment pressure or sump level response **not** consistent with LOCA conditions 3. Ruptured S/G is also faulted outside of containment 4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable steam release from affected S/G to the environment	4. Containment pressure ≥ 60 psig and rising 5. Containment hydrogen concentration ≥ 4% 6. a. Containment pressure ≥ 28 psig **AND** b. **Either** of the following conditions: • < 2 CRFC units operating • < 1 CS pump operating
D **Radiation / Coolant Activity**	3. Containment radiation monitor R-29/R-30 reading > 1.0E+02 R/hr 4. Valid Letdown Monitor (R-9) reading ≥ 24 R/hr 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

NOTES

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.

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.

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

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.

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.

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.**

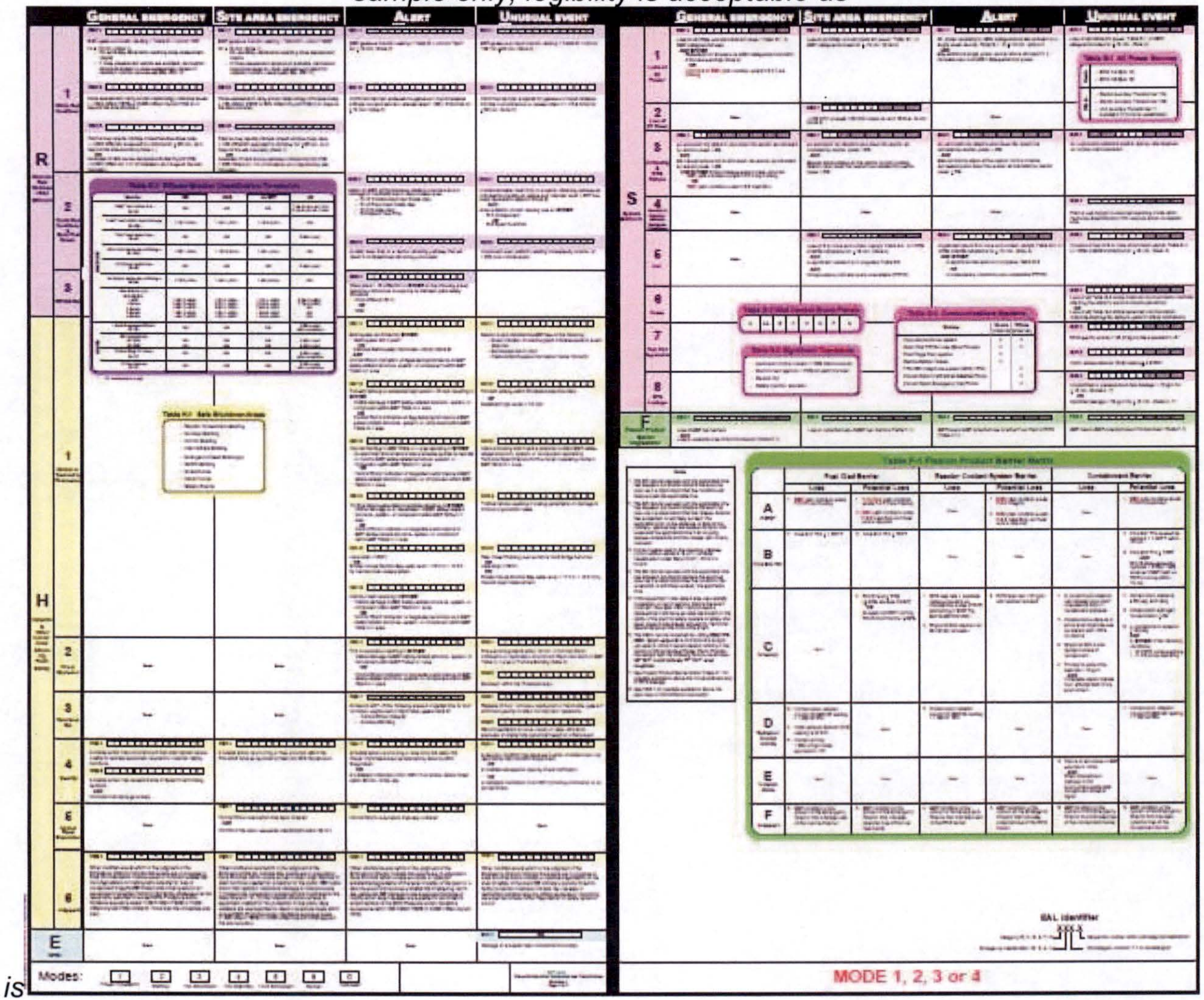
7. See Fission Product Barrier Matrix Table F-1 for possible escalation above the Unusual Event due to RCS leakage.

8. See HS5.1 for possible escalation above the Alert due to Control Room evacuation.

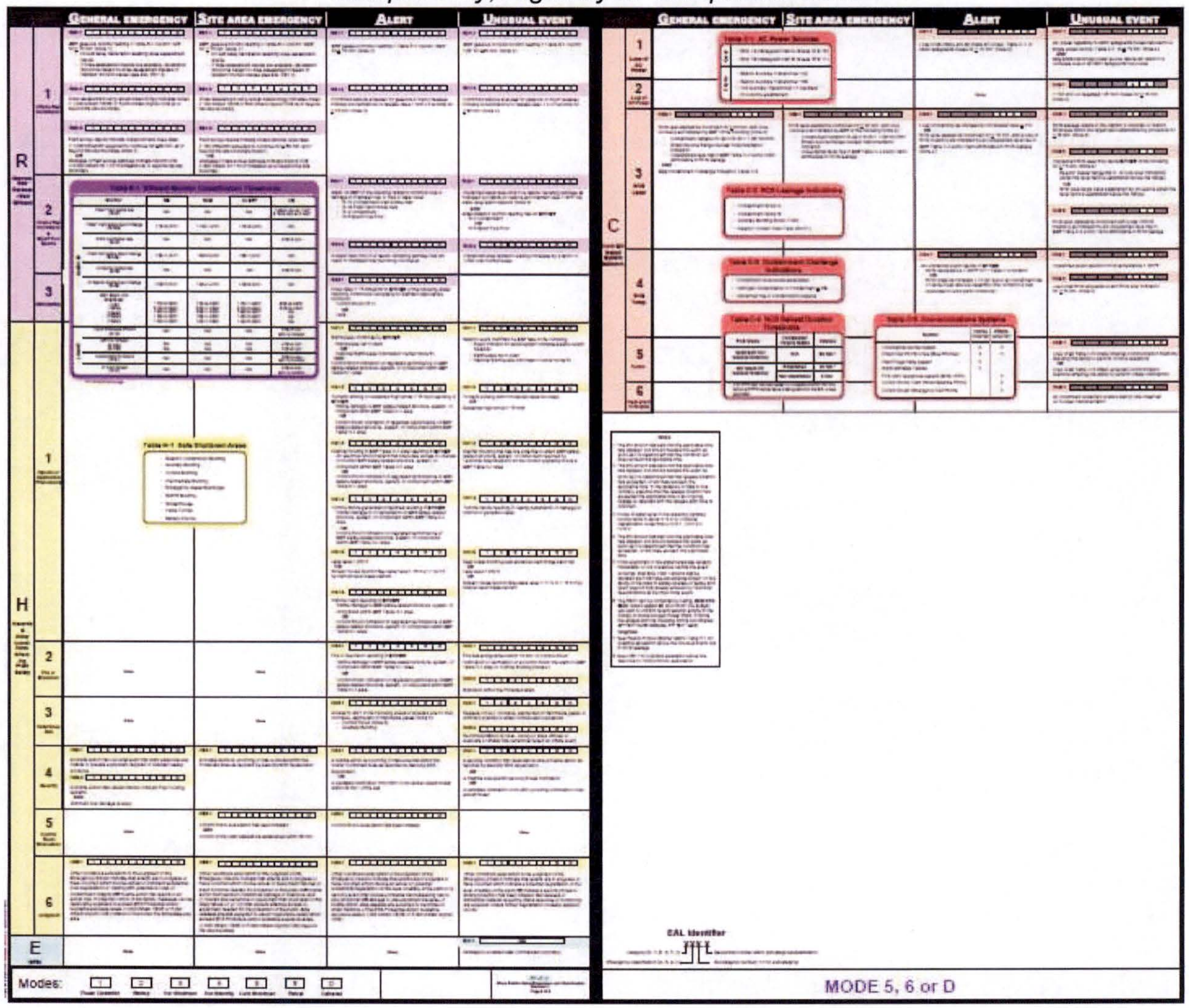
ATTACHMENT 2

EAL WALL CHART (SAMPLE)

Attachment 2
EAL Wall Chart (sample)
sample only, legibility is acceptable as is



Attachment 2
EAL Wall Chart (sample)
sample only, legibility is acceptable as is



ATTACHMENT 3

R. E. GINNA NUCLEAR POWER PLANT EMERGENCY ACTION LEVELS TECHNICAL BASIS

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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 and for explaining event classifications to off-site officials.

The expectation is that emergency classifications are to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes 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
Any Operating Mode:	
R – Abnormal **R**ad Release / Rad Effluent	1 – Offsite Rad Conditions 2 – Onsite Rad Conditions & Spent Fuel Events 3 – CR/CAS/SAS Rad
H – Hazards and Other Conditions Affecting Plant Safety	1 – Natural or Destructive Phenomena 2 – Fire or Explosion 3 – Hazardous Gas 4 – Security 5 – Control Room Evacuation 6 – Judgment
E – ISFSI	None
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – Criticality & RPS Failure 4 – Inability to Reach or Maintain Shutdown Conditions 5 – Instrumentation 6 – Communications 7 – Fuel Clad Degradation 8 – RCS Leakage
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – RCS Level 4 – RCS Temperature 5 – Communications 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:

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

1. First character (letter): Corresponds to the EAL category as described above (R, 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 Wallchart 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 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
CU2.1	CU7	1
CU3.1	CU1	1

GINNA	NEI 99-01	
EAL	IC	Example EAL
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
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
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

GINNA	NEI 99-01	
EAL	IC	Example EAL
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
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

GINNA	NEI 99-01	
EAL	IC	Example EAL
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

SECTION 6.0 - EAL Technical Basis

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. 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	6.54E+6 cpm w/ 1 fan* 4.54E+6 cpm w/ 2 fans*
CNMT Vent Noble Gas Hi Range (R-12A)	1.40E+2 μC/cc	1.40E+1 μC/cc	1.40E+0 μC/cc	N/A
Plant Vent Noble Gas (R-14)	N/A	N/A	N/A	5.80E+5 cpm
Plant Vent Noble Gas Hi Range (R-14A)	1.27E+1 μC/cc	1.27E+0 μC/cc	1.27E-1 μC/cc	N/A
Air Ejector Noble Gas (R-15)	N/A	N/A	N/A	5.30E+5 cpm
Air Ejector Noble Gas Hi Range (R-48)	1.20E+2 μC/cc	1.20E+1 μC/cc	1.20E+0 μC/cc	N/A
Main Steam Line (R-31/R-32)				
1 ARV	1.2E+3 mR/hr	1.2E+2 mR/hr	1.2E+1 mR/hr	8.0E+0 mR/hr
1 Safety	5.7E+2 mR/hr	5.7E+1 mR/hr	5.7E+0 mR/hr	3.7E+0 mR/hr
2 Safety	2.8E+2 mR/hr	2.8E+1 mR/hr	2.8E+0 mR/hr	N/A
3 Safety	1.9E+2 mR/hr	1.9E+1 mR/hr	1.9E+0 mR/hr	N/A
4 Safety	1.4E+2 mR/hr	1.4E+1 mR/hr	1.4E+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

* For containment purge

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 calculated based on annual average X/Q dispersion factors from the ODCM and specified release limits. (ref. 1). These values are discussed in more detail 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. ECP-15-000292-CN-001, CALC-2011-0020 Revision due to SPING Replacement
3. NEI 99-01 AU1
4. EP-EAL-0639, Calculation of Ginna Nuclear Power Station Table R-1 EAL Threshold Values

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 that exceeds 10 mRem TEDE or 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	6.54E+6 cpm w/ 1 fan* 4.54E+6 cpm w/ 2 fans*
CNMT Vent Noble Gas Hi Range (R-12A)	1.40E+2 μC/cc	1.40E+1 μC/cc	1.40E+0 μC/cc	N/A
Plant Vent Noble Gas (R-14)	N/A	N/A	N/A	5.80E+5 cpm
Plant Vent Noble Gas Hi Range (R-14A)	1.27E+1 μC/cc	1.27E+0 μC/cc	1.27E-1 μC/cc	N/A
Air Ejector Noble Gas (R-15)	N/A	N/A	N/A	5.30E+5 cpm
Air Ejector Noble Gas Hi Range (R-48)	1.20E+2 μC/cc	1.20E+1 μC/cc	1.20E+0 μC/cc	N/A
Main Steam Line (R-31/R-32)				
1 ARV	1.2E+3 mR/hr	1.2E+2 mR/hr	1.2E+1 mR/hr	8.0E+0 mR/hr
1 Safety	5.7E+2 mR/hr	5.7E+1 mR/hr	5.7E+0 mR/hr	3.7E+0 mR/hr
2 Safety	2.8E+2 mR/hr	2.8E+1 mR/hr	2.8E+0 mR/hr	N/A
3 Safety	1.9E+2 mR/hr	1.9E+1 mR/hr	1.9E+0 mR/hr	N/A
4 Safety	1.4E+2 mR/hr	1.4E+1 mR/hr	1.4E+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

* For containment purge

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

Monitor indications are calculated (ref. 2) based on annual average X/Q dispersion factors from the ODCM (ref. 1). 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. ECP-15-000292-CN-001, CALC-2011-0020 Revision due to SPING Replacement
3. NEI 99-01 AA1
4. EP-EAL-0639, Calculation of Ginna Nuclear Power Station Table R-1 EAL Threshold Values

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment that exceeds 10 mRem TEDE or 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	6.54E+6 cpm w/ 1 fan* 4.54E+6 cpm w/ 2 fans*
CNMT Vent Noble Gas Hi Range (R-12A)	1.40E+2 μC/cc	1.40E+1 μC/cc	1.40E+0 μC/cc	N/A
Plant Vent Noble Gas (R-14)	N/A	N/A	N/A	5.80E+5 cpm
Plant Vent Noble Gas Hi Range (R-14A)	1.27E+1 μC/cc	1.27E+0 μC/cc	1.27E-1 μC/cc	N/A
Air Ejector Noble Gas (R-15)	N/A	N/A	N/A	5.30E+5 cpm
Air Ejector Noble Gas Hi Range (R-48)	1.20E+2 μC/cc	1.20E+1 μC/cc	1.20E+0 μC/cc	N/A
Main Steam Line (R-31/R-32)				
1 ARV	1.2E+3 mR/hr	1.2E+2 mR/hr	1.2E+1 mR/hr	8.0E+0 mR/hr
1 Safety	5.7E+2 mR/hr	5.7E+1 mR/hr	5.7E+0 mR/hr	3.7E+0 mR/hr
2 Safety	2.8E+2 mR/hr	2.8E+1 mR/hr	2.8E+0 mR/hr	N/A
3 Safety	1.9E+2 mR/hr	1.9E+1 mR/hr	1.9E+0 mR/hr	N/A
4 Safety	1.4E+2 mR/hr	1.4E+1 mR/hr	1.4E+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 9.2E+4 cpm with no isolation

* For containment purge

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 were Calculated in CALC-2011-0020 (ref. 2) based upon a 100 mrem TEDE. The calculations used annual average meteorology from the ODCM (ref. 1). The most restrictive X/Q values at the 0 - 0.5 mile distance were used. EPIP-2-18 (ref. 4) specifies that whole body dose is limiting with respect to emergency classification and protective action recommendations based upon the assumption of a noble gas to iodine ratio of 10,000:1.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. ECP-15-000292-CN-001, CALC-2011-0020 Revision due to SPING Replacement
3. NEI 99-01 AS1
4. EPIP-2-18, Control Room Dose Assessment
5. EP-EAL-0639, Calculation of Ginna Nuclear Power Station Table R-1 EAL Threshold Values

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 (Note 1)

OR

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

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

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	6.54E+6 cpm w/ 1 fan* 4.54E+6 cpm w/ 2 fans*
CNMT Vent Noble Gas Hi Range (R-12A)	1.40E+2 µC/cc	1.40E+1 µC/cc	1.40E+0 µC/cc	N/A
Plant Vent Noble Gas (R-14)	N/A	N/A	N/A	5.80E+5 cpm
Plant Vent Noble Gas Hi Range (R-14A)	1.27E+1 µC/cc	1.27E+0 µC/cc	1.27E-1 µC/cc	N/A
Air Ejector Noble Gas (R-15)	N/A	N/A	N/A	5.30E+5 cpm
Air Ejector Noble Gas Hi Range (R-48)	1.20E+2 µC/cc	1.20E+1 µC/cc	1.20E+0 µC/cc	N/A
Main Steam Line (R-31/R-32)				
1 ARV	1.2E+3 mR/hr	1.2E+2 mR/hr	1.2E+1 mR/hr	8.0E+0 mR/hr
1 Safety	5.7E+2 mR/hr	5.7E+1 mR/hr	5.7E+0 mR/hr	3.7E+0 mR/hr
2 Safety	2.8E+2 mR/hr	2.8E+1 mR/hr	2.8E+0 mR/hr	N/A
3 Safety	1.9E+2 mR/hr	1.9E+1 mR/hr	1.9E+0 mR/hr	N/A
4 Safety	1.4E+2 mR/hr	1.4E+1 mR/hr	1.4E+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

* For containment purge

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 were Calculated in CALC-2011-0020 (ref. 2) based upon a 1000 mrem TEDE. The calculations used annual average meteorology from the ODCM (ref. 1). The most restrictive X/Q values at the 0 - 0.5 mile distance were used. EPIP-2-18 (ref. 4) specifies that whole body dose is limiting with respect to emergency classification and protective action recommendations based upon the assumption of a noble gas to iodine ratio of 10,000:1.

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. ECP-15-000292-CN-001, CALC-2011-0020 Revision due to SPING Replacement
3. NEI 99-01 AG1
4. EPIP-2-18, Control Room Dose Assessment
5. EP-EAL-0639, Calculation of Ginna Nuclear Power Station Table R-1 EAL Threshold Values

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 (Note 1)

OR

Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the site boundary

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

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. Alarm Response Procedure AR-K-29 SFP HI TEMP HI-LO LEVEL
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, 2).

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. 3, 4).

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. ER-SFP.2 Diverse SFP Makeup and Spray
3. Technical Specifications Section 3.9.6 Refueling Cavity Water Level
4. Technical Specifications Section 3.7.11 Spent Fuel Pool (SFP) Water Level
5. 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) • EDG 1B (Safeguard train B, Buses 16 & 17)
Offsite	• Station Auxiliary Transformer 12A • Station Auxiliary Transformer 12B • Unit Auxiliary Transformer 11 backfeed (if currently established)

Mode Applicability:

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

Basis:

Generic

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) • EDG 1B (Safeguard train B, Buses 16 & 17)
Offsite	• Station Auxiliary Transformer 12A • Station Auxiliary Transformer 12B • 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 CA3.1 or EAL CA4.1.

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 CA3.1 or EAL CA4.1.

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):

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

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

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:

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.

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 (including visual observation) 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 $\leq$ 0 in.

OR

RCS level **cannot** be monitored for $\geq$ 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.

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 (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

In the second condition of this EAL, all level indication would be unavailable (including visual observation) 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

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 for ≥ 30 min. with core uncovery as indicated by **ANY** of the following (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:

Generic

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

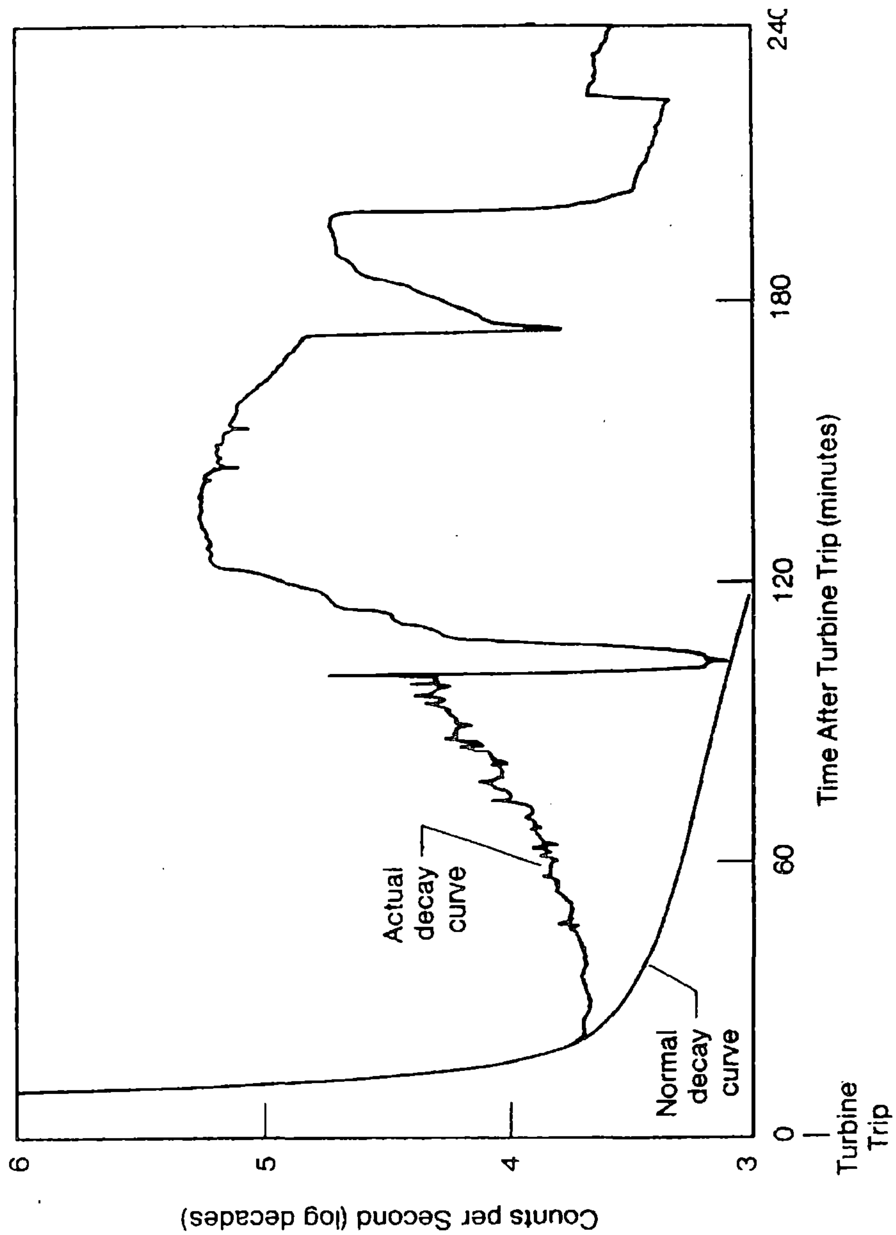
In Refuel or Cold Shutdown mode, normal RCS level indication may be unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor level will not be interrupted. However, RCS level instrumentation is not capable of measuring Reactor Vessel below the bottom of the RCS hotleg (0 in.). If all means of level monitoring are not available, however, the Reactor Vessel inventory loss and possible core uncovery 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 for ≥ 30 min. with core uncovery as indicated by **ANY** of the following (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 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. However, RCS level instrumentation is not capable of measuring Reactor Vessel below the bottom of the RCS hotleg (0 in.) If all means of level monitoring are not available, however, the Reactor Vessel inventory loss and possible core uncovery 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.

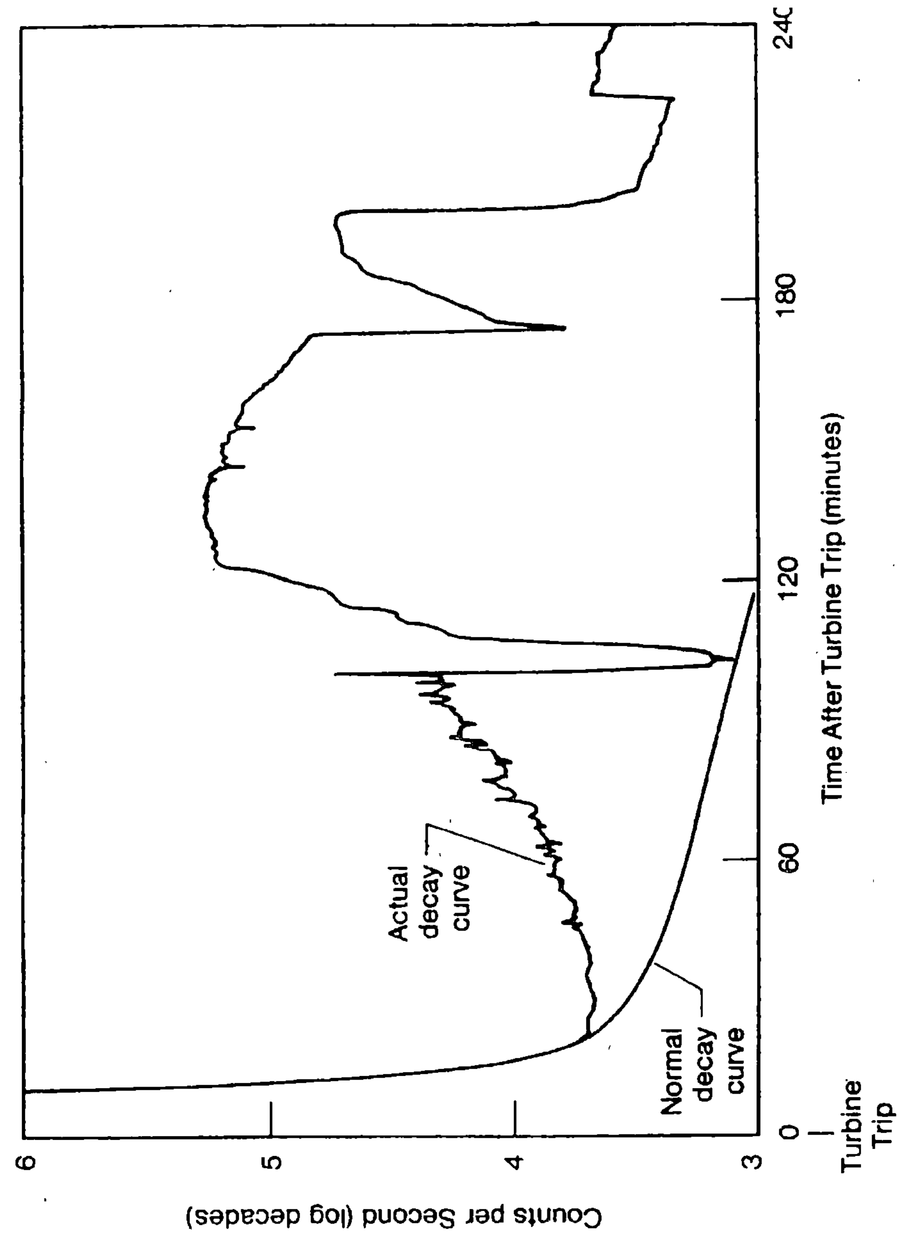
Unplanned

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):

- 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

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

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 CS3.1 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:

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 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-5 (ref. 1, 2). The Direct Dial POTS Lines (Blue Phones) are directly connected to the offsite phone system in area code 315 in Wayne County, independent of the onsite commercial phone system. These phones have both local and long distance direct dialing capability that can support both onsite and offsite communications, including the NRC.

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:

- Event indicator on seismograph indicates seismic event detected
- Earthquake felt in plant
- 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 • Auxiliary Building • Control Building • Intermediate Building • Emergency Diesel Building(s) • SAFW Building • Screenhouse • Cable Tunnel • 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

Earthquake confirmed by **EITHER**:

Earthquake felt in plant

OR

National Earthquake Information Center (Note 6)

AND

Control Room indication of degraded performance of **ANY** safety-related structure, system, or component within **ANY** Table H-1 area

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.**

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 HU1.1 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 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).

Table H-1 Safe Shutdown Areas include all Class 1 Structures and structures containing Class 1 equipment and systems needed for safe shutdown (ref. 5, 6).

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
5. UFSAR Table 3.2-1 Classification of Structures, Systems and Components
6. Ginna Station Fire Protection Program Volume I Part III Section 7.0 Fire Area/Fire Zone Analysis
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:

HA1.2 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 • Auxiliary Building • Control Building • Intermediate Building • Emergency Diesel Building(s) • SAFW Building • Screenhouse • Cable Tunnel • 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 • Auxiliary Building • Control Building • Intermediate Building • Emergency Diesel Building(s) • SAFW Building • Screenhouse • Cable Tunnel • 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
• Auxiliary Building
• Control Building
• Intermediate Building
• Emergency Diesel Building(s)
• SAFW Building
• Screenhouse
• Cable Tunnel
• 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 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

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
• Auxiliary Building
• Control Building
• Intermediate Building
• Emergency Diesel Building(s)
• SAFW Building
• Screenhouse
• Cable Tunnel
• Battery Rooms

Mode Applicability:

All

Basis:

<u>Generic</u>

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, notification is visual observation and report by plant personnel or sensor alarm indication.

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).

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 impeded 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 of the following areas is impeded due to toxic, corrosive, asphyxiant or flammable gases (Note 5):

- Control Room (Note 8)
- Auxiliary Building

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.

Note 8: See HS5.1 for possible escalation above the Alert due to Control Room evacuation.

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 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

Locations designated in the EAL are those areas that are required for plant operation and transitioning to Cold Shutdown that cannot be completed from the Control Room.

Ginna Basis Reference(s):

1. 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 35 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 feed to a steam generator, it was estimated that restoration should be completed in less than 35 minutes. This is consistent with information obtained

during operator walk-throughs of the ER-FIRE procedures which consistently indicated restoration in 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 > 200ºF); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of 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) • EDG 1B (Bus 16)
Offsite	• Station Auxiliary Transformer 12A • Station Auxiliary Transformer 12B • 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) • EDG 1B (Bus 16)
Offsite	• Station Auxiliary Transformer 12A • Station Auxiliary Transformer 12B • Unit Auxiliary Transformer 11 backfeed (if currently established)

Mode Applicability:

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

Basis:

Generic

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 Alert 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 Alert 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) • EDG 1B (Bus 16)
Offsite	• Station Auxiliary Transformer 12A • Station Auxiliary Transformer 12B • 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 a Site Area Emergency 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 (Note 4)

OR

ORANGE or **RED** path condition exists F-0.2 Core Cooling

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) • EDG 1B (Bus 16)
Offsite	• Station Auxiliary Transformer 12A • Station Auxiliary Transformer 12B • 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 reactivity changes. 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 as indicated by reactor power > 5%
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. (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 (< 7%) 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
- 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 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.

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

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 Unusual Event 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.

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 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). The Direct Dial POTS Lines (Blue Phones) are directly connected to the offsite phone system in area code 315 in Wayne County, independent of the onsite commercial phone system.. These phones have both local and long distance direct dialing capability that can support both onsite and offsite communications, including the NRC.

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 during all portions of Modes 1 through 4. 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 ≥ 4.8 R/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 mR/hr ensures timely detection of failed fuel increases greater than 0.1% (ref. 2, 3, 4).

The 4.8 R/hr value for R-9 is based on total RCS activity corresponding to 60 µCi/gm I-131 equivalent. A MicroShield 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. (Notes 4, 7)

OR

Identified leakage > 25 gpm for ≥ 15 min. (Notes 4, 7)

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.

Note 7. See Fission Product Barrier Matrix Table F-1 for possible escalation above the Unusual Event due to RCS leakage.

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 provides 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*

Site Area Emergency:
*Loss or potential loss of **ANY** two barriers*

General Emergency:

*Loss of **ANY** two barriers and loss or potential loss of the third barrier*

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

Section 6.1 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 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 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. 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) 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 2. Containment pressure or sump level response **not** consistent with LOCA conditions 3. Ruptured S/G is also faulted outside of containment 4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable steam release from affected S/G to the environment	4. Containment pressure ≥ 60 psig and rising 5. Containment hydrogen concentration ≥ 4% 6. a Containment pressure ≥ 28 psig **AND** b. **Either** of the following conditions: • < 2 CRFC units operating • < 1 CS pump operating
D Radiation / Coolant Activity	3. Containment radiation monitor R-29/R-30 reading > 1.0E+02 R/hr 4. Valid Letdown Monitor (R-9) reading ≥ 24 R/hr 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

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 $\geq$ 700°F or RVLIS level $\leq$ 52% [55% adverse CNMT]

  OR

- with at least one RCP running RVLIS fluid fraction $\leq$ 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 $\leq$ 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 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 ≥ 700ºF correspond to loss of subcooling.

Plant-Specific

Core Exit Thermocouples (TCs) reading at or in excess of 700ºF 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 700ºF, 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 R/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. The high alarm setting of 200 mR/hr ensures timely detection of failed fuel increases greater than 0.1% (ref. 1). A Letdown Line Monitor reading of 24 R/hr, therefore, 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 ≥ 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

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

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 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).

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
   AND
   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 **AND** Unisolable 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 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

Ginna Basis 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.

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. a Containment pressure ≥ 28 psig **AND** b. **Either** of the following conditions: • < 2 CRFC units operating • < 1 CS pump operating

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 **AND** 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:

Generic

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.

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