NEI 07-01 [Revision 1]

Development of Emergency Action Levels for Passive Reactors

December 2013

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Nuclear Energy Institute

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December 2013

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ACKNOWLEDGMENTS

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EXECUTIVE SUMMARY

Federal regulations require that a nuclear power plant operator develop a scheme for the classification of emergency events and conditions. This scheme is a fundamental component of an emergency plan in that it provides the defined thresholds that will allow site personnel to rapidly implement a range of pre-planned emergency response measures. An emergency classification scheme also facilitates timely decision-making by an Offsite Response Organization (ORO) concerning the implementation of precautionary or protective actions for the public.

The purpose of Nuclear Energy Institute (NEI) 07-01 is to provide guidance to the operators of advanced passive light water reactors for the development of a site-specific emergency classification scheme. The methodology described in this document is consistent with Federal regulations, and related US Nuclear Regulatory Commission (NRC) requirements and guidance. In particular, this methodology has been endorsed by the NRC as an acceptable approach to meeting the requirements of 10 CFR § 50.47(b)(4), related sections of 10 CFR § 50, Appendix E, and the associated planning standard evaluation elements of NUREG-0654/ FEMA-REP-1, Rev. 1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, November 1980.

This revision of NEI 07-01 is based on the material contained in NEI 99-01, *Development of Emergency Action Levels for Non-Passive Reactors*, Revision 6. NEI 99-01, revision 6, represents a significant improvement in emergency classification scheme development guidance, much of which is applicable to the advanced passive reactor designs. NEI 07-01, revision 1, also reflects the more evolved design information currently available for the advanced passive reactor designs (e.g., the Westinghouse AP1000 Design Control Document (DCD).

NEI 07-01 contains a set of generic Initiating Conditions (ICs), Emergency Action Levels (EALs) and fission product barrier status thresholds. It also includes supporting technical basis information, developer notes and recommended classification instructions for users. Users should implement ICs, EALs and thresholds that are as close as possible to the generic material presented in this document with allowance for changes necessary to address site-specific considerations such as plant design, location, terminology, etc.

Properly implemented, the guidance in NEI 07-01 will yield a site-specific emergency classification scheme with clearly defined and readily observable EALs and thresholds. Other benefits include the development of a sound basis document, the adoption of industry-standard instructions for emergency classification (e.g., transient events, classification of multiple events, upgrading, downgrading, etc.), and incorporation of features to improve human performance. An emergency classification using this scheme will be appropriate to the risk posed to plant workers and the public, and should be the same as that made by another NEI 07-01 user plant in response to a similar event.

The individuals responsible for developing an emergency classification scheme are strongly encouraged to review all applicable NRC requirements and guidance prior to beginning their efforts. Questions concerning this document may be directed to the NEI Emergency

Preparedness staff, NEI EAL task force members or submitted to the Emergency Preparedness Frequently Asked Questions process.

Finally, unique State and local requirements associated with an emergency classification scheme are not reflected in this guidance. Incorporation of these requirements may be performed on a case-by-case basis in conjunction with the appropriate ORO agency. Any such changes will require a review under the applicable sections of 10 CFR 50.

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DEVELOMENT OF EMERGENCY ACTION LEVELS FOR PASSIVE REACTORS

1 REGULATORY BACKGROUND

1.1 Operating Reactors

Title 10, Code of Federal Regulations (CFR), Energy, contains the U.S. Nuclear Regulatory Commission (NRC) regulations that apply to nuclear power facilities. Several of these regulations govern various aspects of an emergency classification scheme. A review of the relevant sections listed below will aid the reader in understanding the key terminology provided in Section 3.0 of this document.

- 10 CFR § 50.47(a)(1)(i)
- 10 CFR § 50.47(b)(4)
- 10 CFR § 50.54(q)
- 10 CFR § 50.72(a)
- 10 CFR § 50, Appendix E, IV.B, Assessment Actions
- 10 CFR § 50, Appendix E, IV.C, Activation of Emergency Organization

The above regulations are supplemented by various regulatory guidance documents. Three documents of particular relevance to NEI 99-01 are:

- NUREG-0654/FEMA-REP-1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, October 1980. [Refer to Appendix 1, *Emergency Action Level Guidelines for Nuclear Power Plants*]
- NUREG-1022, *Event Reporting Guidelines 10 CFR § 50.72 and § 50.73*
- Regulatory Guide 1.101, *Emergency Response Planning and Preparedness for Nuclear Power Reactors*

The above list is not all-inclusive and it is strongly recommended that scheme developers consult with licensing/regulatory compliance personnel to identify and understand all applicable requirements and guidance. Questions may also be directed to the NEI Emergency Preparedness staff.

1.2 Independent Spent Fuel Storage Installation (ISFSI)

Selected guidance in NEI 07-01 is applicable to licensees electing to use their 10 CFR 50 emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone ISFSI. The emergency classification levels applicable to an ISFSI are consistent with the requirements of 10 CFR § 50 and the guidance in NUREG 0654/FEMA-REP-1. The

initiating conditions germane to a 10 CFR § 72.32 emergency plan (as described in NUREG-1567) are subsumed within the classification scheme for a 10 CFR § 50.47 emergency plan.

The generic IC and EAL for an ISFSI are presented in Section 8, ISFSI ICs/EALs. IC E-HU1 covers the spectrum of credible natural and man-made events included within the scope of an ISFSI design. This IC is not applicable to installations or facilities that may process and/or repackage spent fuel (e.g., a Monitored Retrievable Storage Facility or an ISFSI at a spent fuel processing facility). In addition, appropriate aspects of IC HU1 and IC HA1 should also be included to address a HOSTILE ACTION directed against an ISFSI.

The analysis of potential onsite and offsite consequences of accidental releases associated with the operation of an ISFSI is contained in NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees*. NUREG-1140 concluded that the postulated worst-case accident involving an ISFSI has insignificant consequences to public health and safety. This evaluation shows that the maximum offsite dose to a member of the public due to an accidental release of radioactive materials would not exceed 1 rem Total Effective Dose Equivalent.

Regarding the above information, the expectations for an offsite response to an Alert classified under a 10 CFR § 72.32 emergency plan are generally consistent with those for a Notification of Unusual Event in a 10 CFR § 50.47 emergency plan (e.g., to provide assistance if requested). Also, the licensee's Emergency Response Organization (ERO) required for 10 CFR § 72.32 emergency plan is different than that prescribed for a 10 CFR § 50.47 emergency plan (e.g., no emergency technical support function).

1.3 NRC Order EA-12-051

The Fukushima Daiichi accident of March 11, 2011, was the result of a tsunami that exceeded the plant's design basis and flooded the site's emergency electrical power supplies and distribution systems. This caused an extended loss of power that severely compromised the key safety functions of core cooling and containment integrity, and ultimately led to core damage in three reactors. While the loss of power also impaired the spent fuel pool cooling function, sufficient water inventory was maintained in the pools to preclude fuel damage from the loss of cooling.

Following a review of the Fukushima Daiichi accident, the NRC concluded that several measures were necessary to ensure adequate protection of public health and safety under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(ii). Among them was to provide each spent fuel pool with reliable level instrumentation to significantly enhance the ability of key decision-makers to allocate resources effectively following a beyond design basis event. To this end, the NRC issued Order EA-12-051, *Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation*, on March 12, 2012, to all US nuclear plants with an operating license, construction permit, or combined construction and operating license.

NRC Order EA-12-051 states, in part, "All licensees … shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification

of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred." To this end, all licensees must provide:

- A primary and back-up level instrument that will monitor water level from the normal level to the top of the used fuel rack in the pool;
- A display in an area accessible following a severe event; and
- Independent electrical power to each instrument channel and provide an alternate remote power connection capability.

NEI 12-02, *Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation"*, provides guidance for complying with NRC Order EA-12-051.

NEI 07-01, Revision 1, includes three EALs that reflect the availability of the enhanced spent fuel pool level instrumentation associated with NRC Order EA-12-051. These EALs are included within existing IC AA2, and new ICs AS2 and AG2. Associated EAL notes, bases and developer notes are also provided.

1.4 APPLICABILITY TO SMALL MODULAR REACTOR DESIGNS

There are significant design and operating differences between large nuclear power plants and Small Modular Reactors (SMRs) (e.g., differences in source term). For this reason, this document is not applicable to SMRs.

2 KEY TERMINOLOGY USED IN NEI 07-01

There are several key terms that appear throughout the NEI 07-01 methodology. These terms are introduced in this section to support understanding of subsequent material. As an aid to the reader, the following table is provided as an overview to illustrate the relationship of the terms to each other.

Emergency Classification Level			
Unusual Event	Alert	SAE	GE
↓	↓	↓	↓
Initiating Condition	Initiating Condition	Initiating Condition	Initiating Condition
↓	↓	↓	↓
Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis
(1) - When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition. This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.			

2.1 EMERGENCY CLASSIFICATION LEVEL (ECL)

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

- Notification of Unusual Event (NOUE)
- Alert

- Site Area Emergency (SAE)
- General Emergency (GE)

2.1.1 Notification of Unusual Event (NOUE)[1]

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.

Purpose: The purpose of this classification is to assure that the first step in future response has been carried out, to bring the operations staff to a state of readiness, and to provide systematic handling of unusual event information and decision-making.

2.1.2 Alert

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

Purpose: The purpose of this classification is to assure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required, and provide offsite authorities current information on plant status and parameters.

2.1.3 Site Area Emergency

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

Purpose: The purpose of the Site Area Emergency declaration is to assure that emergency response centers are staffed, to assure that monitoring teams are dispatched, to assure that personnel required for evacuation of near-site areas are at duty stations if the situation becomes more serious, to provide consultation with offsite authorities, and to provide updates to the public through government authorities.

2.1.4 General Emergency (GE)

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

[1] This term is sometimes shortened to Unusual Event (UE) or other similar site-specific terminology. The terms Notification of Unusual Event, NOUE and Unusual Event are used interchangeably throughout this document.

ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Purpose: The purpose of the General Emergency declaration is to initiate predetermined protective actions for the public, to provide continuous assessment of information from the licensee and offsite organizational measurements, to initiate additional measures as indicated by actual or potential releases, to provide consultation with offsite authorities, and to provide updates for the public through government authorities.

2.2 INITIATING CONDITION (IC)

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

Discussion: An IC describes an event or condition, the severity or consequences of which meets the definition of an emergency classification level. An IC can be expressed as a continuous, measurable parameter (e.g., RCS leakage), an event (e.g., an earthquake) or the status of one or more fission product barriers (e.g., loss of the RCS barrier).

Appendix 1 of NUREG-0654 does not contain example Emergency Action Levels (EALs) for each ECL, but rather Initiating Conditions (i.e., plant conditions that indicate that a radiological emergency, or events that could lead to a radiological emergency, has occurred). NUREG-0654 states that the Initiating Conditions form the basis for establishment by a licensee of the specific plant instrumentation readings (as applicable) which, if exceeded, would initiate the emergency classification. Thus, it is the specific instrument readings that would be the EALs.

Considerations for the assignment of a particular Initiating Condition to an emergency classification level are discussed in Section 3.

2.3 EMERGENCY ACTION LEVEL (EAL)

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

Discussion: EAL statements may utilize a variety of criteria including instrument readings and status indications; observable events; results of calculations and analyses; entry into particular procedures; and the occurrence of natural phenomena.

2.4 FISSION PRODUCT BARRIER THRESHOLD

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

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

- Fuel Clad
- Reactor Coolant System (RCS)
- Containment

Upon determination that one or more fission product barrier thresholds have been exceeded, the combination of barrier loss and/or potential loss thresholds is compared to the fission product barrier IC/EAL criteria to determine the appropriate ECL.

In some accident sequences, the ICs and EALs presented in the Abnormal Radiation Levels/ Radiological Effluent (A) Recognition Category will be exceeded at the same time, or shortly after, the loss of one or more fission product barriers. This redundancy is intentional as the former ICs address radioactivity releases that result in certain offsite doses from whatever cause, including events that might not be fully encompassed by fission product barriers (e.g., spent fuel pool accidents, design containment leakage following a LOCA, etc.).

3 DESIGN OF THE NEI 07-01 EMERGENCY CLASSIFICATION SCHEME

3.1 ASSIGNMENT OF EMERGENCY CLASSIFICATION LEVELS (ECLS)

An effective emergency classification scheme must incorporate a realistic and accurate assessment of risk, both to plant workers and the public. There are obvious health and safety risks in underestimating the potential or actual threat from an event or condition; however, there are also risks in overestimating the threat as well (e.g., harm that may occur during an evacuation). The NEI 07-01 emergency classification scheme attempts to strike an appropriate balance between reasonably anticipated event or condition consequences, potential accident trajectories, and risk avoidance or minimization.

There are a range of "non-emergency events" reported to the US Nuclear Regulatory Commission (NRC) staff in accordance with the requirements of 10 CFR § 50.72. Guidance concerning these reporting requirements, and example events, are provided in NUREG-1022. Certain events reportable under the provisions of 10 CFR § 50.72 may also require the declaration of an emergency.

In order to align each Initiating Conditions (IC) with the appropriate ECL, it was necessary to determine the attributes of each ECL. The goal of this process is to answer the question, "What events or conditions should be placed under each ECL?" The following sources provided information and context for the development of ECL attributes.

- Assessments of the effects and consequences of different types of events and conditions
- Typical abnormal and emergency operating procedure setpoints and transition criteria
- Typical Technical Specification limits and controls
- Offsite Dose Calculation Manual (ODCM) radiological release limits
- Review of selected Safety Analysis Report (SAR) accident analyses
- Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs)
- NEI 99-01, *Development of Emergency Action Levels for Non-Passive Reactors*, Revision 6
- Industry Operating Experience
- Input from industry subject matter experts and NRC staff members

The following ECL attributes were created by the Revision 1 Preparers to aid in the development of ICs and Emergency Action Levels (EALs). The preparers decided to include the attributes in this revision since they may be useful in briefing and training settings (e.g., helping an Emergency Director understand why a particular condition is classified as an Alert). It should be stressed that developers not attempt to redefine these

attributes or apply them in any fashion that would change the generic guidance contained in this document[2].

The attributes of each ECL are presented below.

3.1.1 Notification of Unusual Event (NOUE)

A Notification of Unusual Event, as defined in section 2.1.1, includes but is not limited to an event or condition that involves:

(A) A precursor to a more significant event or condition.

(B) A minor loss of control of radioactive materials or the ability to control radiation levels within the plant.

(C) A consequence otherwise significant enough to warrant notification to local, State and Federal authorities.

3.1.2 Alert

An Alert, as defined in section 2.1.2, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of either the fuel clad or Reactor Coolant System (RCS) fission product barrier.

(B) An event or condition that significantly reduces the margin to a loss or potential loss of the fuel clad or RCS fission product barrier.

(C) A significant loss of control of radioactive materials resulting in an inability to control radiation levels within the plant, or a release of radioactive materials to the environment that could result in doses greater than 1% of an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION occurring within the OWNER CONTROLLED AREA, including those directed at an Independent Spent Fuel Storage Installation (ISFSI).

3.1.3 Site Area Emergency

A Site Area Emergency, as defined in section 2.1.3, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of any two fission product barriers - fuel clad, RCS and/or containment.

[2] The use of ECL attributes is at the discretion of a licensee and is not a requirement of the NRC. If a licensee chooses to incorporate the ECL attributes into their scheme basis document, it must be very clear that the NRC staff has not endorsed their acceptability or application for any purpose. In particular, the staff does not consider the attribute statements to supersede the established ECL definitions. As a result, the use of the attributes as a basis for justifying EAL changes is unacceptable.

(B) A precursor event or condition that may lead to the loss or potential loss of multiple fission product barriers within a relatively short period of time. Precursor events and conditions of this type include those that challenge the monitoring and/or control of multiple safety systems.

(C) A release of radioactive materials to the environment that could result in doses greater than 10% of an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION occurring within the plant PROTECTED AREA.

3.1.4 General Emergency

A General Emergency, as defined in section 2.1.4, includes but is not limited to an event or condition that involves:

(A) Loss of any two fission product barriers AND loss or potential loss of the third barrier - fuel clad, RCS and/or containment.

(B) A precursor event or condition that, unmitigated, may lead to a loss of all three fission product barriers. Precursor events and conditions of this type include those that lead directly to core damage and loss of containment integrity.

(C) A release of radioactive materials to the environment that could result in doses greater than an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION resulting in the loss of key safety functions (reactivity control, core cooling/RPV water level or RCS heat removal) or damage to spent fuel.

3.1.5 Risk-Informed Insights

Emergency preparedness is a defense-in-depth measure that is independent of the assessed risk from any particular accident sequence; however, the development of an effective emergency classification scheme can benefit from a review of risk-based assessment results. To that end, the development and assignment of certain ICs and EALs also considered insights from several site-specific probabilistic safety assessments (PSA - also known as probabilistic risk assessment, PRA). Some generic insights from this review included:

1. Accident sequences involving a prolonged loss of all required DC power are significant contributors to core damage frequency. For this reason, a loss of all required DC power for greater than 15 minutes, with the plant at or above Safe Shutdown, was assigned an ECL of Site Area Emergency. Precursor events to a loss of all DC power were also included as an Unusual Event and an Alert.

2. Advanced passive light water reactor PRAs indicate that both core damage frequency (CDF) and large release frequency (LRF) are less than that for non-advanced passive reactors. However, the generic methodology still needs to be sufficiently rigorous to address sequences involving containment bypass, large Loss of Coolant Accidents (LOCA) with early containment failure, anticipated transients without scram (ATWS), and extended loss of electrical power (DC and AC).

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

The NEI 07-01 methodology makes use of symptom-based, barrier-based and event-based ICs and EALs. Each type is discussed below.

Symptom-based ICs and EALs are parameters or conditions that are measurable over some range using plant instrumentation (e.g., core temperature, reactor coolant level, radiological effluent, etc.). When one or more of these parameters or conditions are off-normal, reactor operators will implement procedures to control relevant plant parameters and ensure safe operating conditions.

Fission product barrier-based ICs and EALs are the subset of symptom-based EALs that refer specifically to the level of challenge to the principal barriers against the release of radioactive material from the reactor core to the environment. These barriers are the fuel cladding, the reactor coolant system pressure boundary, and the containment. The barrier-based ICs and EALs consider the level of challenge to each individual barrier - potentially lost and lost - and the total number of barriers under challenge.

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. These include the failure of an automatic reactor scram/trip to shut down the reactor, natural phenomena (e.g., an earthquake), or man-made hazards such as a toxic gas release.

3.3 NSSS DESIGN DIFFERENCES

The NEI 07-01 emergency classification scheme accounts for the design differences between the currently certified advanced passive PWR (AP1000) and BWR (ESBWR) designs by specifying EALs, where appropriate, unique to each type of Nuclear Steam Supply System (NSSS).

Developers will need to consider the relevant aspects of their plant's design and operating characteristics when converting the generic guidance of this document into a site-specific classification scheme. The goal is to maintain as much fidelity as possible to the intent of generic ICs and EALs within the constraints imposed by the plant design and operating characteristics.

The guidance in NEI 07-01 is not applicable to advanced non-passive light water reactor designs such as the Advanced Boiling Water Reactor (ABWR), the Advanced Pressurized Water Reactor (APWR), and the Evolutionary Power Reactor (EPR). An Emergency Classification Scheme for these designs should be developed in accordance with NEI 99-01, *Development of Emergency Action Levels for Non-Passive Reactors*.

3.4 ORGANIZATION AND PRESENTATION OF GENERIC INFORMATION

The scheme's generic information is organized by Recognition Category in the following order.

- A - Abnormal Radiation Levels / Radiological Effluent – Section 6
- C - Cold Shutdown / Refueling System Malfunction – Section 7

- E - Independent Spent Fuel Storage Installation (ISFSI) – Section 8
- F - Fission Product Barrier – Section 9
- H - Hazards and Other Conditions Affecting Plant Safety – Section 10
- S - System Malfunction – Section 11

Each Recognition Category section contains a matrix showing the ICs and their associated emergency classification levels.

The following information and guidance is provided for each IC:

- **ECL** – the assigned emergency classification level for the IC.
- **Initiating Condition** – provides a summary description of the emergency event or condition.
- **Operating Mode Applicability** – Lists the modes during which the IC and associated EAL(s) are applicable (i.e., are to be used to classify events or conditions).
- **Example Emergency Action Level(s)** – Provides examples of reports and indications that are considered to meet the intent of the IC. Developers should address each example EAL. If the generic approach to the development of an example EAL cannot be used (e.g., an assumed instrumentation range is not available at the plant), the developer should attempt to specify an alternate means for identifying entry into the IC.

  For Recognition Category F, the fission product barrier thresholds are presented in tables applicable to the ESBWR and AP1000, and arranged by fission product barrier and the degree of barrier challenge (i.e., potential loss or loss). This presentation method shows the synergism among the thresholds, and supports accurate assessments.
- **Basis** – Provides background information that explains the intent and application of the IC and EALs. In some cases, the basis also includes relevant source information and references.
- **Developer Notes** – Information that supports the development of the site-specific ICs and EALs. This may include clarifications, references, examples, instructions for calculations, etc. Developer notes should not be included in the site's emergency classification scheme basis document. Developers may elect to include information resulting from a developer note action in a basis section.
- **ECL Assignment Attributes** – Located within the Developer Notes section, specifies the attribute used for assigning the IC to a given ECL.

3.5 IC AND EAL MODE APPLICABILITY

The NEI 07-01 emergency classification scheme was developed recognizing that the applicability of ICs and EALs will vary with plant mode. For example, some symptom-based ICs and EALs can be assessed only during the power operations, startup, hot shutdown/hot standby or safe shutdown modes of operation when all fission product barriers are in place, and plant instrumentation and safety systems are fully operational. In the cold shutdown and refueling modes, different symptom-based ICs and EALs will come into play to reflect the opening of systems for routine maintenance, the unavailability of some safety system components and the use of alternate instrumentation.

Developers will need to incorporate the mode criteria from unit-specific Technical Specifications into their emergency classification scheme. The following table shows which Recognition Categories are applicable in each plant mode. The ICs and EALs for a given Recognition Category are applicable in the indicated modes.

MODE APPLICABILITY MATRIX						
	Recognition Category					
Mode	**A**	**C**	**E**	**F**	**H**	**S**
Power Operation	X		X	X	X	X
Startup	X		X	X	X	X
Hot Shutdown/Hot Standby	X		X	X	X	X
Safe Shutdown	X		X	X	X	X
Cold Shutdown	X	X	X		X	
Refueling	X	X	X		X	
Defueled	X	X	X		X	

ESBWR Operating Modes	
Power Operation (1):	Mode Switch in Run
Startup (2):	Mode Switch in Startup or Refuel (with all vessel head bolts fully tensioned)
Hot Shutdown (3):	Mode Switch in Shutdown, Average Reactor Coolant Temperature > 420°F
Safe Shutdown (4)	Mode Switch in Shutdown, Average Reactor Coolant Temperature ≤ 420°F and > 200°F
Cold Shutdown (5):	Mode Switch in Shutdown, Average Reactor Coolant Temperature ≤ 200°F
Refueling (6):	Mode Switch in Shutdown or Refuel, and one or more vessel head bolts less than fully tensioned.

AP1000 Operating Modes	
Power Operation (1):	Reactor Power > 5%, Keff ≥ 0.99
Startup (2):	Reactor Power ≤ 5%, Keff ≥ 0.99
Hot Standby (3):	RCS > 420°F, Keff < 0.99
Safe Shutdown (4):	200°F < RCS ≤ 420°F, Keff < 0.99
Cold Shutdown (5):	RCS ≤ 200°F, Keff < 0.99
Refueling (6):	One or more vessel head closure bolts less than fully tensioned

The scheme must also include the following mode designation specific to NEI 07-01:

Defueled (None):	All fuel removed from the reactor vessel (i.e., full core offload during refueling or extended outage).

4 SITE-SPECIFIC SCHEME DEVELOPMENT GUIDANCE

This section provides detailed guidance for developing a site-specific emergency classification scheme. Conceptually, the approach discussed here mirrors the approach used to prepare emergency operating procedures – generic material prepared by reactor vendor owners groups is converted by each nuclear power plant into site-specific emergency operating procedures. Likewise, the emergency classification scheme developer will use the generic guidance in NEI 07-01 to prepare a site-specific emergency classification scheme and the associated basis document.

It is important that the NEI 07-01 emergency classification scheme be implemented as an integrated package. Selected use of portions of this guidance is strongly discouraged as it will lead to an inconsistent or incomplete emergency classification scheme that will likely not receive the necessary regulatory approval.

4.1 GENERAL IMPLEMENTATION GUIDANCE

The guidance in NEI 07-01 is not intended to be applied to plants "as-is"; however, developers should attempt to keep their site-specific schemes as close to the generic guidance as possible. The goal is to meet the intent of the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) within the context of site-specific characteristics – locale, plant design, operating features, terminology, etc. Meeting this goal will result in a shorter and less cumbersome NRC review and approval process, closer alignment with the schemes of other nuclear power plant sites and better positioning to adopt future industry-wide scheme enhancements.

When properly developed, the ICs and EALs should be unambiguous and readily assessable.

As discussed in Section 3, the generic guidance includes ICs and example EALs. It is the intent of this guidance that both be included in site-specific documents as each serves a specific purpose. The IC is the fundamental event or condition requiring a declaration. The EAL(s) is the pre-determined threshold that defines when the IC is met. If some feature of the plant location or design is not compatible with a generic IC or EAL, efforts should be made to identify an alternate IC or EAL.

If an IC or EAL includes an explicit reference to a mode dependent technical specification limit that is not applicable to the plant, then that IC and/or EAL need not be included in the site-specific scheme. In these cases, developers must provide adequate documentation to justify why the IC and/or EAL are not incorporated (i.e., sufficient detail to allow a third party to understand the decision not to incorporate the generic guidance).

Useful acronyms and abbreviations associated with the NEI 07-01 emergency classification scheme are presented in Appendix A, Acronyms and Abbreviations. Site-specific entries may be added if necessary. Those acronyms and abbreviations not used in the site-specific scheme can be deleted.

Many words or terms used in the NEI 07-01 emergency classification scheme have scheme-specific definitions. These words and terms are set in all capital letters (i.e., ALL CAPS). The definitions are presented in Appendix B, Definitions.

Below are examples of acceptable modifications to the generic guidance. These may be incorporated depending upon site developer and user preferences.

- The ICs within a Recognition Category may be placed in reverse order for presentation purposes (e.g., start with a General Emergency at the left/top of a user aid, followed by Site Area Emergency, Alert and NOUE).

- The Initiating Condition numbering may be changed.

- The first letter of a Recognition Category designation may be changed, as follows, provided the change is carried through for all of the associated IC identifiers.

    - R may be used in lieu of A

    - M may be used in lieu of S

For example, the Abnormal Radiation Levels / Radiological Effluent category designator “A” (for Abnormal) may be changed to “R” (for Radiation). This means that the associated ICs would be changed to RU1, RU2, RA1, etc.

- The ICs and EALs from Recognition Categories S and C may be incorporated into a common presentation method (e.g., one table) provided that all related notes and mode applicability requirements are maintained.

- The ICs and EALs for Emergency Director judgment and security-related events may be placed under separate Recognition Categories.

- The terms EAL and threshold may be used interchangeably.

The material in the Developer Notes section is included to assist developers with crafting correct IC and EAL statements. This material is not required to be in the final emergency classification scheme basis document.

4.2 CRITICAL CHARACTERISTICS

As discussed above, developers are encouraged to keep their site-specific schemes as close to the generic guidance as possible. When crafting the scheme, developers should satisfy themselves that certain critical characteristics have been met. These critical characteristics are listed below.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are consistent with industry guidance; while the actual wording may be different, the classification intent is maintained. With respect to Recognition Category F, a site-specific scheme must include some type of user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. The user-aid logic must be consistent with the classification logic presented in Section 9.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are technically complete and accurate (i.e., they contain the information necessary to make a correct classification).
- EAL statements use objective criteria and observable values.
- ICs, EALs, Operating Mode Applicability and Note statements and formatting consider human factors and are user-friendly.
- The scheme facilitates upgrading and downgrading of the emergency classification where necessary.
- The scheme facilitates classification of multiple concurrent events or conditions.

4.3 INSTRUMENTATION USED FOR EALS

Instrumentation referenced in EAL statements should include that described in the emergency plan section which addresses 10 CFR 50.47(b)(8) and (9) and/or the SAR. Instrumentation used for EALs need not be safety-related, addressed by a Technical Specification or ODCM control requirement, nor powered from an emergency power source; however, EAL developers should strive to incorporate instrumentation that is reliable and routinely maintained in accordance with site programs and procedures. Alarms referenced in EAL statements should be those that are the most operationally significant for the described event or condition.

Scheme developers should ensure that specified values used as EAL setpoints are within the calibrated range of the referenced instrumentation, and consider any automatic instrumentation functions that may impact accurate EAL assessment. In addition, EAL setpoint values should not use terms such as “off-scale low” or “off-scale high” since that type of reading may not be readily differentiated from an instrument failure. Findings and violations related to EAL instrumentation issues may be located on the NRC website.

4.4 PRESENTATION OF SCHEME INFORMATION TO USERS

The US Nuclear Regulatory Commission (NRC) expects licensees to establish and maintain the capability to assess, classify and declare an emergency condition promptly within 15 minutes after the availability of indications to plant operators that an emergency action level has been, or may be, exceeded. When writing an emergency classification procedure and creating related user aids, the developer must determine the presentation method(s) that best supports the end users by facilitating accurate and timely emergency classification. To this end, developers should consider the following points.

- The first users of an emergency classification procedure are the operators in the Control Room. During the allowable classification time period, they may have responsibility to perform other critical tasks, and will likely have minimal assistance in making a classification assessment.
- As an emergency situation evolves, members of the Control Room staff are likely to be the first personnel to notice a change in plant conditions. They can assess the changed conditions and, when warranted, recommend a different emergency

classification level to the Technical Support Center (TSC) and/or Emergency Operations Facility (EOF).

- Emergency Directors in the TSC and/or EOF will have more opportunity to focus on making an emergency classification, and will probably have advisors from Operations available to help them.

Emergency classification scheme information for end users should be presented in a manner with which licensed operators are most comfortable. Developers will need to work closely with representatives from the Operations and Operations Training Departments to develop readily usable and easily understood classification tools (e.g., a procedure and related user aids). If necessary, an alternate method for presenting emergency classification scheme information may be developed for use by Emergency Directors and/or Offsite Response Organization personnel.

A wallboard is an acceptable presentation method provided that it contains all the information necessary to make a correct emergency classification. This information includes the ICs, Operating Mode Applicability criteria, EALs and Notes. Notes may be kept with each applicable EAL or moved to a common area and referenced; a reference to a Note is acceptable as long as the information is adequately captured on the wallboard and pointed to by each applicable EAL[3]. Basis information need not be included on a wallboard but it should be readily available to emergency classification decision-makers.

In some cases, it may be advantageous to develop two wallboards - one for use during power operations, startup and hot conditions, and another for cold shutdown and refueling conditions.

Alternative presentation methods for the Recognition Category F ICs and fission product barrier thresholds are acceptable and include flow charts, block diagrams, and checklist-type tables. Developers must ensure that the site-specific method addresses all possible threshold combinations and classification outcomes shown in the ESBWR or AP1000 EAL fission product barrier tables. The NRC staff considers the presentation method of the Recognition Category F information to be an important user aid and may request a change to a particular proposed method if, among other reasons, the change is necessary to promote consistency across the industry.

4.5 INTEGRATION OF ICS/EALS WITH PLANT PROCEDURES

A rigorous integration of IC and EAL references into plant operating procedures is not recommended. This approach would greatly increase the administrative controls and workload for maintaining procedures. On the other hand, performance challenges may occur if recognition of meeting an IC or EAL is based solely on the memory of a licensed operator or an Emergency Director, especially during periods of high stress.

[3] Where appropriate, the Notes shown in the generic guidance typically include the event/condition ECL and the duration time specified in the EAL. If developers prefer to have several ICs reference a common NOTE on a wallboard display, it is acceptable to remove the ECL and time criterion and use a generic statement. For example, a common NOTE could read "The Emergency Director should declare the emergency promptly upon determining that the applicable EAL time has been exceeded, or will likely be exceeded."

Developers should consider placing appropriate visual cues (e.g., a step, note, caution, etc.) in plant procedures alerting the reader/user to consult the site emergency classification procedure. Visual cues could be placed in emergency operating procedures, abnormal operating procedures, alarm response procedures, and normal operating procedures that apply to cold shutdown and refueling modes. As an example, a step, note or caution could be placed at the beginning of an RCS leak abnormal operating procedure that reminds the reader that an emergency classification assessment should be performed.

4.6 BASIS DOCUMENT

A basis document is an integral part of an emergency classification scheme. The material in this document supports proper emergency classification decision-making by providing informing background and development information in a readily accessible format. It can be referred to in training situations and when making an actual emergency classification, if necessary. The document is also useful for establishing configuration management controls for EP-related equipment and explaining an emergency classification to offsite authorities. The content of the basis document should include, at a minimum, the following:

- A site-specific Mode Applicability Matrix and description of operating modes, similar to that presented in section 3.5.

- A discussion of the emergency classification and declaration process reflecting the material presented in Section 5. This material may be edited as needed to align with site-specific emergency plan and implementing procedure requirements.

- Each Initiating Condition along with the associated EALs or fission product barrier thresholds, Operating Mode Applicability, Notes and Basis information.

- A listing of acronyms and defined terms, similar to that presented in Appendices A and B, respectively. This material may be edited as needed to align with site-specific characteristics.

- Any site-specific background or technical appendices that the developers believe would be useful in explaining or using elements of the emergency classification scheme.

A Basis section should not contain information that could modify the meaning or intent of the associated IC or EAL. Such information should be incorporated within the IC or EAL statements, or as an EAL Note. Information in the Basis should only clarify decision-making for an emergency classification.

Basis information should be readily available to be referenced, if necessary, by the Emergency Director. For example, a copy of the basis document could be maintained in the appropriate emergency response facilities.

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Director refers to it during an event), the NRC

staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q).

4.7 EAL/Threshold References to Abnormal Operating Procedure (AOP) and Emergency Operating Procedure (EOP) Setpoints/Criteria

As reflected in the generic guidance, the criteria/values used in several EALs and fission product barrier thresholds may be drawn from plant-specific AOPs and EOPs. This approach is intended to maintain good alignment between operational diagnoses and emergency classification assessments. Developers should verify that appropriate administrative controls are in place to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required.

4.8 Developer and User Feedback

Questions or comments concerning the material in this document may be directed to the NEI Emergency Preparedness staff, NEI EAL task force members or submitted to the Emergency Preparedness Frequently Asked Questions process.

5 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

5.1 GENERAL CONSIDERATIONS

When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.

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

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

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

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

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.); the EAL and/or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be

exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

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

5.2 Classification Methodology

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL(s) must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, then the IC is considered met and the associated ECL is declared in accordance with plant procedures.

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

5.3 Classification of Multiple Events and Conditions

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

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

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

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

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

5.4 Consideration of Mode Changes During Classification

The mode in effect at the time that an event or condition occurred, and prior to any plant

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

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Safe Shutdown/Stable Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Safe Shutdown [ESBWR/AP1000] mode or higher.

5.5 Classification of Imminent Conditions

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

5.6 Emergency Classification Level Upgrading and Downgrading

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

The following approach to downgrading or terminating an ECL is recommended.

ECL	Action When Condition No Longer Exists
Unusual Event	Terminate the emergency in accordance with plant procedures.
Alert	Downgrade or terminate the emergency in accordance with plant procedures.
Site Area Emergency with no long-term plant damage	Downgrade or terminate the emergency in accordance with plant procedures.
Site Area Emergency with long-term plant damage	Terminate the emergency and enter recovery in accordance with plant procedures.
General Emergency	Terminate the emergency and enter recovery in accordance with plant procedures.

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

5.7 Classification of Short-Lived Events

As discussed in Section 3.2, event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include a failure of the reactor protection system to automatically scram/trip the reactor followed by a successful manual scram/trip or an earthquake.

5.8 Classification of Transient Conditions

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

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

<u>EAL momentarily met but the condition is corrected prior to an emergency declaration</u> – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example (AP1000):

> An ATWS occurs and the normal feedwater system fails. Steam generator levels rapidly decrease and the plant enters an inadequate RCS heat removal condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts the startup feedwater system in accordance with an EOP step and clears the inadequate RCS heat removal condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event; emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations where an operator is able to take a successful corrective action prior to the Emergency Director completing the review and steps necessary to make the emergency declaration.

This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

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

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

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

5.10 Retraction of an Emergency Declaration

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

6 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

Table A-1: Recognition Category "A" Initiating Condition Matrix

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
AU1 Release of gaseous or liquid radioactivity greater than 2 times the Off-site Dose Calculation Manual limits for 60 minutes or longer. *Op. Modes: All*	**AA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE. *Op. Modes: All*	**AS1** Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE. *Op. Modes: All*	**AG1** Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE. *Op. Modes: All*
AU2 UNPLANNED loss of water level above irradiated fuel. *Op. Modes: All*	**AA2** Significant lowering of water level above, or damage to, irradiated fuel. *Op. Modes: All*	**AS2** Spent fuel pool level at (site-specific Level 3 description). *Op. Modes: All*	**AG2** Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer. *Op. Modes: All*
	AA3 Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown. *Op. Modes: All*		

Table intended for use by EAL developers. Inclusion in licensee documents is not required.

AU1

ECL: Notification of Unusual Event

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the Off-site Dose Calculation Manual limits for 60 minutes or longer.

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2 or 3)

Notes:

- The Emergency Director should declare the Unusual Event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

(1) Reading on **ANY** effluent radiation monitor greater than 2 times the Off-site Dose Calculation Manual limits for 60 minutes or longer:

(site-specific monitor list and threshold values corresponding to 2 times the ODCM limits)

(2) Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the Off-site Dose Calculation Manual limits for 60 minutes or longer.

Basis:

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

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

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

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

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

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

EAL #1 - This EAL addresses radioactivity releases from continuously monitored gaseous or liquid effluent pathways.

EAL #2 - This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL will typically be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

EAL #3 - This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC AA1.

Developer Notes:

The Offsite Dose Calculation Manual (ODCM) implements regulations related to effluent controls (e.g., 10 CFR Part 20 and 10 CFR Part 50, Appendix I). The ODCM methodology should be used for establishing the monitor thresholds for this IC.

Listed monitors should include the effluent monitors described in the ODCM.

Developers may also consider including installed monitors associated with other potential effluent pathways that are not described in the ODCM[4,5]. If included, EAL values for these monitors should be determined using the most applicable dose/release limits presented in the ODCM. It is recognized that a calculated EAL value may be below what the monitor can read; in that case, the monitor does not need to be included in the list. Also, some monitors may not be governed by Technical Specifications or other license-related related requirements; therefore, it is important that the associated EAL and basis section clearly identify any limitations on the use or availability of these monitors.

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

Radiation monitor readings should reflect values that correspond to a radiological release

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[4] This includes consideration of the effluent monitors described in the site emergency plan section(s) which address the requirements of 10 CFR 50.47(b)(8) and (9).

[5] Developers should keep in mind the requirements of 10 CFR 50.54(q) and the guidance provided by INPO related to emergency response equipment when considering the addition of other effluent monitors.

exceeding 2 times the ODCM limit. The ODCM describes methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values. In cases where a methodology is not adequately defined, developers should determine values consistent with effluent control regulations (e.g., 10 CFR Part 20 and 10 CFR Part 50 Appendix I) and related guidance.

For EAL #2 - Values in this EAL should be 2 times the setpoint established by the radioactivity discharge permit to warn of a release that is not in compliance with the specified limits. Indexing the value in this manner ensures consistency between the EAL and the setpoint established by a specific discharge permit.

Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).

It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.

Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.

ECL Assignment Attributes: 3.1.1.B

AU2

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following:

(site-specific level indications).

AND

b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors.

(site-specific list of area radiation monitors)

Basis:

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

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

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

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

Escalation of the emergency classification level would be via IC AA2.

Developer Notes:

The "site-specific level indications" are those indications that may be used to monitor water level in the various portions of the REFUELING PATHWAY. Specify the mode applicability of a particular indication if it is not available in all modes.

The "site-specific list of area radiation monitors" should contain those area radiation monitors that would be expected to have increased readings following a decrease in water level in the site-specific REFUELING PATHWAY. In cases where a radiation monitor(s) is not available or would not provide a useful indication, consideration should be given to including alternate indications such as UNPLANNED changes in tank and/or sump levels.

Development of the EALs should consider the availability and limitations of mode-dependent, or other controlled but temporary, radiation monitors. Specify the mode applicability of a particular monitor if it is not available in all modes.

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B

AA1

ECL: Alert

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

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2 or 3 or 4)

Notes:

- The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:

(site-specific monitor list and threshold values)

(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).

(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.

(4) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point):

- Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

Basis:

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

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Developer Notes:

While this IC may not be met absent challenges to one or more fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Updated Final Safety Analysis Report, the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.

The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE.…".

The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.

The "site-specific monitor list and threshold values" should be determined with consideration of the following:

- Selection of the appropriate installed gaseous and liquid effluent monitors.
- The effluent monitor readings should correspond to a dose of 10 mrem TEDE or 50 mrem

thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

- Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AS1 and AG1. Acceptable sources of this information include, but are not limited to, the ODCM and values used in the site's emergency dose assessment methodology.

- The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AS1 and AG1. Acceptable sources of this information include, but are not limited to, the ODCM and values used in the site's emergency dose assessment methodology.

- Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.

The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.

Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).

It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.

Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.

Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.

ECL Assignment Attributes: 3.1.2.C

AA2

ECL: Alert

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

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2 or 3)

(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.

(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the following radiation monitors:

(site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)

(3) Lowering of spent fuel pool level to (site-specific Level 2 value). [*See Developer Notes*]

Basis:

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

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.

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

<u>EAL #1</u>

This EAL escalates from AU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.

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

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

<u>EAL #2</u>

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

<u>EAL #3</u>

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

Escalation of the emergency classification level would be via ICs AS1 or AS2 (*see AS2 Developer Notes*).

Developer Notes:

For EAL #1

Depending upon the availability and range of instrumentation, this EAL may include specific readings indicative of fuel uncovery; consider water and radiation level readings. Specify the mode applicability of a particular indication if it is not available in all modes.

For EAL #2

The "site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms" should contain those radiation monitors that could be used to identify damage to an irradiated fuel assembly (e.g., confirmatory of a release of fission product gases from irradiated fuel).

For EALs #1 and #2

Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).

It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may

choose not to include the monitor as an indication and identify an alternate EAL threshold.

To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.

Development of the EALs should also consider the availability and limitations of mode-dependent, or other controlled but temporary, radiation monitors. Specify the mode applicability of a particular monitor if it is not available in all modes.

For EAL #3

The "site-specific Level 2 value" is usually the spent fuel pool level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owners' group guidance.

Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 2 value.

ECL Assignment Attributes: 3.1.2.B and 3.1.2.C

AA3

ECL: Alert

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

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2)

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

(1) Dose rate greater than 15 mR/hr in **ANY** of the following areas:

- Control Room
- Central Alarm Station
- (other site-specific areas/rooms)

(2) An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas:

(site-specific list of plant rooms or areas with entry-related mode applicability identified)

Basis:

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

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

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase

occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.

- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

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

Developer Notes:

<u>EAL #1</u>

The value of 15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times.

The "other site-specific areas/rooms" should include any areas or rooms requiring <u>continuous</u> occupancy to maintain normal plant operation, or to perform a normal cooldown and shutdown.

For identified areas that do not have permanently installed area radiation monitoring, this EAL will have to be assessed by local survey.

<u>EAL #2</u>

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

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

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

Rooms and areas listed in EAL #1 do not need to be included in EAL #2, including the Control Room.

ECL Assignment Attributes: 3.1.2.C

AS1

ECL: Site Area Emergency

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

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2 or 3)

Notes:

- The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:

(site-specific monitor list and threshold values)

(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point).

(3) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point):

- Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.

Basis:

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

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

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

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

Escalation of the emergency classification level would be via IC AG1.

Developer Notes:

While this IC may not be met absent challenges to multiple fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Design Control Document (DCD), the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.

The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE.…".

The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.

The "site-specific monitor list and threshold values" should be determined with consideration of the following:

- Selection of the appropriate installed gaseous effluent monitors.
- The effluent monitor readings should correspond to a dose of 100 mrem TEDE or 500 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.
- Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AA1 and AG1. Acceptable sources of this information include, but are not limited to, the ODCM and values used in the site's emergency dose assessment methodology.

- The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AA1 and AG1. Acceptable sources of this information include, but are not limited to, the ODCM and values used in the site's emergency dose assessment methodology.

- Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.

The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.

Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).

It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.

Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.

Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.

ECL Assignment Attributes: 3.1.3.C

AS2

[*See Developer Notes*]

ECL: Site Area Emergency

Initiating Condition: Spent fuel pool level at (site-specific Level 3 description)

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Lowering of spent fuel pool level to (site-specific Level 3 value).

Basis:

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

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

Escalation of the emergency classification level would be via IC AG1 or AG2.

Developer Notes:

In accordance with the discussion in Section 1.4, NRC Order EA-12-051, it is recommended that this IC and EAL be implemented when the enhanced spent fuel pool level instrumentation is available for use. The “site-specific Level 3 value” is usually that spent fuel pool level where fuel remains covered and actions to implement make-up water addition should no longer be deferred. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owner’s group guidance.

Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 3 value.

ECL Assignment Attributes: 3.1.3.B

AG1

ECL: General Emergency

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

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2 or 3)

Notes:

- The Emergency Director should declare the General Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:

(site-specific monitor list and threshold values)

(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point).

(3) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point):

- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.

Basis:

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

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

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

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

Developer Notes:

The effluent ICs/EALs are included to provide a basis for classifying events that cannot be readily classified on the basis of plant conditions alone. The inclusion of both types of ICs/EALs more fully addresses the spectrum of possible events and accidents.

While this IC may not be met absent challenges to multiple fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Design Control Document (DCD), the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.

The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE.…".

The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.

The "site-specific monitor list and threshold values" should be determined with consideration of the following:

- Selection of the appropriate installed gaseous effluent monitors.
- The effluent monitor readings should correspond to a dose of 1,000 mrem TEDE or 5,000 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.
- Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AA1 and AS1. Acceptable sources of this information include, but are not limited to, the ODCM and values used in the site's emergency dose assessment methodology.

- The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AA1 and AS1. Acceptable sources of this information include, but are not limited to, the ODCM and values used in the site's emergency dose assessment methodology.

- Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.

The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.

Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).

It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.

Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.

Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.

ECL Assignment Attributes: 3.1.4.C

AG2

[*See Developer Notes*]

ECL: General Emergency

Initiating Condition: Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer

Operating Mode Applicability: All

Example Emergency Action Levels:

Note: The Emergency Director should declare the General Emergency promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.

(1) Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer.

Basis:

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

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

Developer Notes:

In accordance with the discussion in Section 1.4, NRC Order EA-12-051, it is recommended that this IC and EAL be implemented when the enhanced spent fuel pool level instrumentation is available for use. The "site-specific Level 3 value" is usually that spent fuel pool level where fuel remains covered and actions to implement make-up water addition should no longer be deferred. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owner's group guidance.

Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 3 value.

ECL Assignment Attributes: 3.1.4.C

7 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

Table C-1: Recognition Category "C" Initiating Condition Matrix

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
CU1 Inability to restore and maintain required (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory level. *Op. Modes: Cold Shutdown, Refueling*	**CA1** Loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory. *Op. Modes: Cold Shutdown, Refueling*	**CS1** Loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory affecting core decay heat removal capability. *Op. Modes: Cold Shutdown, Refueling*	**CG1** Loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory affecting fuel clad integrity with containment challenged. *Op. Modes: Cold Shutdown, Refueling*
CU2 UNPLANNED increase in RCS temperature. *Op. Modes: Cold Shutdown, Refueling*	**CA2** Inability to maintain the plant in cold shutdown. *Op. Modes: Cold Shutdown, Refueling*		
CU3 Loss of all ability to charge at least one (site-specific Class 1E battery) for 30 minutes or longer. *Op. Modes: Cold Shutdown, Refueling*	**CA3** Loss of all required (site-specific Class 1E DC power) or (site-specific Class 1E UPS bus power) for 15 minutes or longer. *Op. Modes: Cold Shutdown, Refueling*		
	CA4 Extended loss of all offsite and all onsite AC power to (site-specific buses) *Op. Modes: Cold Shutdown, Refueling, Defueled*		
CU5 Loss of all onsite or offsite communications capabilities. *Op. Modes: Cold* Shutdown, *Refueling, Defueled*			

Table intended for use by EAL developers. Inclusion in licensee documents is not required.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
CU6 UNPLANNED partial loss of monitoring or control functions for 15 minutes or longer *Op. Modes: Cold* Shutdown*, Refueling, Defueled*			
	CA7 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. *Op. Modes: Cold Shutdown, Refueling*		

Table intended for use by EAL developers. Inclusion in licensee documents is not required.

CU1

ECL: Notification of Unusual Event

Initiating Condition: Inability to restore and maintain required (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory level

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Inability to restore and maintain (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level greater than a required lower limit for 15 minutes or longer.

(2) a. (Reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level cannot be monitored.

AND

b. UNPLANNED increase in (site-specific sump and/or tank) levels due to loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory.

Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

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

EAL #1 recognizes that the minimum required (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

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

EAL #2 addresses a condition where all means to determine (reactor vessel/RCS [*AP1000R*] or RPV [*ESBWR*]) level have been lost. All means include local and remote instrumentation, cameras and direct observation. In this condition, operators may determine that an inventory loss

is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]).

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

Developer Notes:

EAL #1 – It is recognized that the minimum allowable reactor vessel/RCS/RPV level may have many values over the course of a refueling outage. Developers should solicit input from licensed operators concerning the optimum wording for this EAL statement. In particular, determine if the generic wording is adequate to ensure accurate and timely classification, or if specific setpoints can be included without making the EAL statement unwieldy or potentially inconsistent with actions that may be taken during an outage. If specific setpoints are included, these should be drawn from applicable operating procedures or other controlling documents.

EAL #2.b – Enter any "site-specific sump and/or tank" levels that could be expected to increase if there were a loss of inventory (i.e., the lost inventory would enter the listed sump or tank). These should include, but are not necessarily limited to, Containment Sumps [AP1000] and Drywell and Equipment Drain Sumps [ESBWR].

ECL Assignment Attributes: 3.1.1.A

CU2

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED increase in RCS temperature

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) UNPLANNED increase in RCS temperature to greater than 200 ºF.

(2) Loss of **ALL** RCS temperature and (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level indication for 15 minutes or longer.

Basis:

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or the inability to determine RCS temperature and level, represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA2.

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

EAL #1 involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

EAL #2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA2 based on exceeding plant configuration-specific time criteria.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

CU3

ECL: Notification of Unusual Event

Initiating Condition: Loss of all ability to charge at least one (site-specific Class 1E DC battery) for 30 minutes or longer

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

(1) Loss of all ability to charge at least one (site-specific Class 1E 24-hr. DC battery) for 30 minutes or longer.

Basis:

The Class 1E 24-hr. (safety-related) DC system provides electrical power for safety-related and vital control and monitoring instrumentation loads. It also provides power for safe shutdown when all the on-site and off-site AC power sources are lost and cannot be recovered for 72 hours. The offsite AC power system normally supplies power for the unit in cold shutdown, refueling, and defueled conditions. Both the normal offsite and standby onsite AC power systems are non-Class 1E with no Technical Specification requirements. All safety-related functions associated with the unit in cold shutdown and refueling are provided by the safety-related onsite Class 1E DC power systems including the 120V Vital AC power system supplied from the batteries powering the safety-related DC buses through inverters. The Passive ALWRs also have standby diesel generators that are not safety-related. Storage batteries are the safety-related power source for Class 1E electric power. However, this non-Class 1E AC power is required to charge the safety-related DC batteries and maintain long-term safety-related DC bus voltage.

AP1000

The loss of normal offsite AC power and standby onsite AC power systems de-energizes the RNS pumps. Therefore, the progression of events after a loss of RNS cooling at mid-loop caused by a loss of AC power results in a heatup, an eventual boiling off of coolant, reduction of hot leg level, and actuation of passive IRWST injection. This restores RCS water inventory using only the passive cooling systems and the onsite safety-related Class 1E 24-hr. DC power systems.

ESBWR

The loss of normal offsite AC power and standby AC power systems de-energizes the RWCU/SDC pumps. The onsite Class 1E DC power system is rated for 72 hours of service based on the instrumentation and control power for systems required for safe shutdown, and thus remains available for a significant time following a loss of all offsite and onsite AC power.

30 minutes was chosen to allow sufficient time for plant personnel to attempt to establish a viable offsite or diesel generator AC power supply to the (site-specific) AC power buses required to charge one or more Class 1E 24-hr. DC batteries.

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA2, or an IC in Recognition Category A.

Developer Notes:

Buses ECS-ES-1 and 2 for AP1000 or the PIP A3 and B3 buses for ESBWR are the source of non-Class 1E AC power to the Class 1E DC battery chargers.

ECL Assignment Attributes: 3.1.1.A

CU5

ECL: Notification of Unusual Event

Initiating Condition: Loss of all onsite or offsite communications capabilities

Operating Mode Applicability: Cold Shutdown, Refueling, Defueled

Example Emergency Action Levels: (1 or 2 or 3)

(1) Loss of **ALL** of the following onsite communication methods:

(site-specific list of communications methods)

(2) Loss of **ALL** of the following ORO communications methods:

(site-specific list of communications methods)

(3) Loss of **ALL** of the following NRC communications methods:

(site-specific list of communications methods)

Basis:

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

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

EAL #1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL #2 addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are (see Developer Notes).

EAL #3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

Developer Notes:

EAL #1 - The “site-specific list of communications methods” should include all communications methods used for routine plant communications (e.g., commercial or site telephones, page-party systems, radios, etc.). This listing should include installed plant equipment and components, and not items owned and maintained by individuals.

EAL #2 - The “site-specific list of communications methods” should include all communications methods used to perform initial emergency notifications to OROs as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. Example methods are ring-down/dedicated telephone lines, commercial telephone lines, radios, satellite telephones and internet-based communications technology.

In the Basis section, insert the site-specific listing of the OROs requiring notification of an emergency declaration from the Control Room in accordance with the site Emergency Plan, and typically within 15 minutes.

EAL #3 – The “site-specific list of communications methods” should include all communications methods used to perform initial emergency notifications to the NRC as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. These methods are typically the dedicated Emergency Notification System (ENS) telephone line and commercial telephone lines.

ECL Assignment Attributes: 3.1.1.C

CU6

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED partial loss of monitoring or control functions for 15 minutes or longer

Operating Mode Applicability: Cold Shutdown, Refueling, Defueled

Example Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) UNPLANNED loss of the ability to monitor or control one or more of the following key safety functions from the Main Control Room required for the current plant operating mode for 15 minutes or longer:

- Decay Heat Removal
- (RCS Inventory [*AP1000*]/ RPV Level [*ESBWR*]) Control
- Reactivity Control
- Spent Fuel Pool Level/Temperature Control

Basis:

This IC recognizes the difficulty associated with monitoring and controlling changing plant conditions without the use of a major portion of the control and indication systems. A Notification of Unusual Event level is considered appropriate for this loss of monitoring and control IC due to the inherently safer condition of the core when in the cold condition. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems in the Main Control Room due to an inadvertent loss.

<u>AP1000</u>

Systems that provide monitoring and control capability include:

- The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shut down the plant, and to maintain the plant in a safe shutdown condition.
- The Plant Control System (PLS) is a non-safety-related system and includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine.

- The Data Display and Processing System (DDS) (Plant Computer) consists of a set of graphics workstations that obtain their inputs from real-time data networks and deliver their output to the network for plant operators and other users.

- The Diverse Actuation System (DAS) is a non-safety-related system available to ensure monitoring and control capability as well as providing an alternative means of initiating a reactor trip and selected engineering safety features.

ESBWR

The Essential Distributed Control and Information System (E-DCIS) provides redundant data communications networks to support the monitoring and control of interfacing safety-related control and instrumentation systems. The system includes electrical devices and circuitry that connect field sensors, display devices, controllers, power supplies, and actuators, which are part of these safety-related systems. The E-DCIS also includes any associated data acquisition and communications software, if required, to support its distribution function of data and control. The system processes data from safety-related systems and safety-related trip or initiation data strictly through E-DCIS, while non-safety-related data is processed through the Non-Essential DCIS

The E-DCIS provides the data processing and transmission network that encompasses the four independent and separate data multiplexing divisions 1, 2, 3, and 4, corresponding to the four divisions of safety-related electrical and I&C equipment.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

CA1

ECL: Alert

Initiating Condition: Loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the Alert promptly upon determining that the specified time limit has been exceeded, or will likely be exceeded.

(1) (Reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level less than (site-specific level) for 15 minutes or longer.

(2) a. (Reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level cannot be monitored for 30 minutes or longer

AND

b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory.

Basis:

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

For EAL #1, a lowering of water level below (site-specific level) for 15 minutes or longer indicates that operator actions have not been successful in restoring and maintaining (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, EAL #1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a residual heat removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

If the (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory level continues to lower below the TAF, then escalation to Site Area Emergency would be via IC CS1.

For EAL #2, the inability to monitor (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]).

The 30-minute duration for the loss of level indication was chosen to allow CA1 to be an effective precursor to CS1. This provides time to increase makeup and isolate leakage prior to core uncovery. Whether or not the actions in progress will be effective should be apparent within 30 minutes. When in Cold Shutdown or Refueling the event can be classified as an Alert due to the significantly reduced decay heat and lower temperature and pressure. This increases the time available to resolve the problem. Significant fuel damage is not expected to occur until after core uncovery has occurred per the analysis referenced in the CG1 basis.

If (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory level monitoring capability is not restored within 60 minutes with indications that the core may be uncovered, then escalation to Site Area Emergency would be via IC CS1.

Developer Notes:

For EAL #1 – the "site-specific level" should be based on:

<u>AP1000</u>

For both Cold Shutdown and Refueling modes RCS hot leg level below the Level 1 Hot Leg Level actuation setpoint for greater than 15 minutes should be used to indicate RCS makeup efforts are failing. This setpoint actuates ADS-4 and IRWST injection.

<u>ESBWR</u>

For both Cold Shutdown and Refueling modes RPV below the RPV Level 0.5 actuation setpoint (1 m above TAF) for greater than 15 minutes should be used to indicate that RCS makeup efforts are failing. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier. This setpoint actuates GDCS equalization and is below the setpoint that GDCS injection should have initiated.

For EAL #2 - The type and range of RCS level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for the AP1000. As appropriate to the plant design, alternate means of determining RCS level may be installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. The instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.

Enter any "site-specific sump and/or tank" levels that could be expected to increase if there were a loss of inventory (i.e., the lost inventory would enter the listed sump or tank). These should include, but are not necessarily limited to, Containment Sumps [AP1000] and Drywell and Equipment Drain Sumps [ESBWR].

ECL Assignment Attributes: 3.1.2.B

CA2

ECL: Alert

Initiating Condition: Inability to maintain the plant in cold shutdown

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

(1) UNPLANNED increase in RCS temperature to greater than 200 ºF for greater than the duration specified in the following table.

Table: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up Duration**
Intact (but not at reduced inventory [*AP1000*])	Not applicable	60 minutes*
Not intact (or at reduced inventory [*AP1000*])	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

(2) RCS pressure increase due to UNPLANNED RCS heatup greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [*AP1000*])

Basis:

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

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit [200 ºF] when the heat removal function is available does not warrant a classification.

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

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute

time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact or is at reduced inventory [*AP1000*], and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the Containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

EAL #2 provides a pressure-based indication of RCS heat-up.

Escalation of the emergency classification level would be via IC CS1 or AS1.

Developer Notes:

For EAL #1 –The RCS should be considered intact or not intact in accordance with site-specific criteria. Assessment of RCS temperature for this threshold should be consistent with instrumentation used to determine RCS temperature relative to the Technical Specification requirements.

For EAL #2 - The "site-specific pressure reading" should be the lowest change in pressure that can be accurately determined using installed instrumentation, but not less than 10 psig.

For AP1000s, this IC and its associated EALs address the concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*. A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show that there are sequences that can cause core uncovery in 15 to 20 minutes, and severe core damage within an hour after decay heat removal is lost. The allowed time frames are consistent with the guidance provided by Generic Letter 88-17 and believed to be conservative given that a low pressure Containment barrier to fission product release is established.

ECL Assignment Attributes: 3.1.2.B

CA3

ECL: Alert

Initiating Condition: Loss of all required (site-specific Class 1E DC power) or (site-specific Class 1E UPS bus power) for 15 minutes or longer

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Indicated voltage is less than (site-specific bus voltage value) on (ALL required site-specific Class 1E 24-hr. DC buses) for 15 minutes or longer.

(2) Loss of power to ALL (site-specific Class 1E UPS buses) for 15 minutes or longer.

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of the Class 1E 24-hr. DC, which provides electrical power for safety-related and vital control and monitoring instrumentation loads. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

Routine maintenance on a division related basis is performed during shutdown periods. It is intended that the loss of the operating (operable) division is to be considered. If this loss results in the inability to maintain cold shutdown, then the Alert criteria of CA2 - Inability to Maintain Plant in Cold Shutdown would also be met in addition to the CA3 Alert criteria.

Threshold 1 (site-specific minimum bus voltage) is the minimum bus voltage necessary for the operation of safety-related instrumentation and controls. This voltage value incorporates a margin significantly longer than the allowed 15 minutes of operation before the onset of inability to operate those loads.

Threshold 2 addresses an event that results in de-energizing all Class 1E UPS busses. This condition would result in degraded capability to monitor and control the unit from the Main Control Room resulting in the need for additional personnel to manage the event.

Developer Notes:

The site-specific Class 1E DC bus voltage should be based on the minimum battery terminal voltage at the end of the discharge period.

The site-specific Class 1E UPS buses are the 120 VAC safety-related buses.

ECL Assignment Attributes: 3.1.2.B

CA4

ECL: Alert

Initiating Condition: Extended loss of all offsite and all onsite AC power to (site-specific buses)

Operating Mode Applicability: Cold Shutdown, Refueling, Defueled

Example Emergency Action Levels:

Note: The Emergency Director should declare the Alert promptly upon determining that the specified time limit has been exceeded, or will likely be exceeded.

(1) Loss of **ALL** offsite and **ALL** onsite AC Power to (site-specific buses) for greater than 2 hours

Basis:

Loss of all AC power compromises all plant systems requiring electric power including heat removal, Spent Fuel heat removal and the active inventory makeup systems.

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 buses.

In the event of total loss of offsite or onsite standby AC power sources, the system stationary batteries shall constitute the source of electric power for operation of the Non-Class 1E DC and UPS loads for at least 2 hours. Therefore, the time selected as the threshold allows additional time for recovery of AC power prior to escalation to Alert.

Developer Notes:

The "site-specific buses" are the non-Class 1E buses fed by either offsite or onsite AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. For the AP1000, these are the ECS-ES-1 and ECS-ES-2 buses. For the ESBWR, these are the PIP buses A3 and B3.

ECL Assignment Attributes: 3.1.2.B

CA7

ECL: Alert

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

Operating Mode Applicability: Cold Shutdown, Refueling, Defueled

Example Emergency Action Levels:

(1) a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- (site-specific hazards)
- Other events with similar hazard characteristics as determined by the Shift Manager

AND

b. **EITHER** of the following:

1. Event damage has caused indications of degraded performance in at least one division of a SAFETY SYSTEM needed for the current operating mode

   OR

2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Basis:

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

Condition 1.b.1 addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be

significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Condition 1.b.2 addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC CS1 or AS1.

Developer Notes:

For (site-specific hazards), developers should consider adding other significant, site-specific hazards to the list contained in EAL 1.a (e.g., a seiche or tsunami).

Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant divisions of equipment in accordance with site-specific design criteria.

ECL Assignment Attributes: 3.1.2.B

CS1

ECL: Site Area Emergency

Initiating Condition: Loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory affecting core decay heat removal capability

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

(1) RPV level less than (site-specific level corresponding to TAF) for 30 minutes or longer [*ESBWR*].

(2) a. (Reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **ANY** of the following:

- (Site-specific radiation monitor) reading greater than (site-specific value)
- Erratic source range monitor indication [*AP1000*]
- (Other site-specific indications)

Basis:

This IC addresses a significant and prolonged loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory control and makeup capability leading to core uncovery and IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

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

In EAL 2.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). The passive cooling systems should continue to function and provide a sufficient volume of water for cooling, therefore, 30-minute duration allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that core uncovery has occurred by observing the alternative indications specified.

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

Escalation of the emergency classification level would be via IC CG1 or AG1.

Developer Notes:

The type and range of RCS/reactor vessel/RPV level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for an AP1000. As appropriate to the plant design, alternate means of determining RCS/reactor vessel/RPV level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. Additionally, for the AP1000, the instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.

<u>AP1000</u>

The basis for this IC and associated example EALs is reactor vessel/RCS level below the TAF for great than 30 min. However, the lowest observable level on AP1000 RCS level instrumentation is the lowest observable level on the RCS Hot Leg Level Instrument. Therefore this example EAL is not applicable to the AP1000 design. Once RCS level has dropped below the minimum observable level, classification is made based on EAL #2.

For EAL #2.b – first bullet - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a “site-specific radiation monitor” that could be used to detect core uncovery and the associated “site-specific value” indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.

For EAL #2.b – second bullet - Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this

should be used as a tool for making such determinations. Developers should verify that plant-specific source range monitor design supports this alternative core uncovery indicator.

For EAL #2.b – third bullet - Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.

ESBWR

For EAL #1 – The “site-specific level” should be that which corresponds to the top of active fuel.

For EAL #2.b – first bullet - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a “site-specific radiation monitor” that could be used to detect core uncovery and the associated “site-specific value” indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.

If installed radiation monitors are not capable of indicating core uncovery, alternate site-specific level indications of core uncovery should be used if available.

For EAL #2.b – second bullet - Because ESBWR source range monitor (SRM) nuclear instrumentation detectors are typically located below core mid-plane, this may not be a viable indicator of core uncovery for ESBWRs.

For EAL #2.b – third bullet - Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.

ECL Assignment Attributes: 3.1.3.B

CG1

ECL: General Emergency

Initiating Condition: Loss of (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) inventory affecting fuel clad integrity with containment challenged

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the General Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

(1) a. RPV level less than (site-specific level corresponding to TAF) for 30 minutes or longer [*ESBWR*].

AND

b. **ANY** indication from the Containment Challenge Table (see below).

(2) a. (Reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **ANY** of the following:

- (Site-specific radiation monitor) reading greater than (site-specific value)
- Erratic source range monitor indication [*AP1000*]
- (Other site-specific indications)

AND

c. **ANY** indication from the Containment Challenge Table (see below).

Containment Challenge Table
■ CONTAINMENT CLOSURE not established*
■ (Explosive mixture) exists inside containment
■ UNPLANNED increase in containment pressure
■ Reactor Building radiation monitor reading above (site-specific value) [*ESBWR*]

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

Basis:

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

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

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

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

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

An unplanned increase in containment pressure may be an indication of energy addition to the containment (RCS heatup or boiling) under conditions when the containment barrier has limited pressure retention capability challenging containment closure.

In EAL 2, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). The passive cooling systems should continue to function and provide a sufficient volume of water for cooling, therefore, 30-minute duration allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor (reactor vessel/RCS [*AP1000*] or RPV [*ESBWR*]) level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that core uncovery has occurred by observing the alternative indications specified.

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*;

SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*.

Developer Notes:

The type and range of RCS level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for an AP1000. As appropriate to the plant design, alternate means of determining RCS level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. Additionally, for the AP1000, the instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.

AP1000

The basis for this IC and associated example EALs is reactor vessel/RCS level below the TAF for great than 30 min in combination with indications of a containment challenge. However, the lowest observable level on AP1000 RCS level instrumentation is the lowest observable level on the RCS Hot Leg Level Instrument. Therefore this example EAL is not applicable to the AP1000 design. Once RCS level has gone below that minimum observable level, classification is made based on EAL #2.

For EAL #2.b - first bullet - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a “site-specific radiation monitor” that could be used to detect core uncovery and the associated “site-specific value” indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.

For EAL #2.b – second bullet - Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations. Developers should verify that plant-specific source range monitor design supports this alternative core uncovery indicator.

For EAL #2.b – third bullet - Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.

ESBWR

For EAL #1 – The "site-specific level" should be that which corresponds to the top of active fuel.

For EAL #2.b – first bullet - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site-specific value" indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

For EAL #2.b - second bullet - Because ESBWR Source Range Monitor (SRM) nuclear instrumentation detectors are typically located below core mid-plane, this may not be a viable indicator of core uncovery for ESBWRs.

For EAL #2.b – third bullet - Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.

For the Containment Challenge Table:

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

For "Explosive mixture", developers may enter the minimum containment atmospheric hydrogen concentration necessary to support a hydrogen burn (i.e., the lower deflagration limit). A concurrent containment oxygen concentration may be included if the plant has this indication available in the Control Room.

For ESBWRs, the use of Reactor Building radiation monitors should provide indication of increased release that may be indicative of a challenge to secondary containment. The "site-specific value" should be based on the EOP maximum safe values because these values are easily recognizable and have a defined basis.

ECL Assignment Attributes: 3.1.4.B

8 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

Table E-1: Recognition Category "E" Initiating Condition Matrix

UNUSUAL EVENT
E-HU1 Damage to a loaded cask CONFINEMENT BOUNDARY. *Op. Modes: All*

Table intended for use by EAL developers. Inclusion in licensee documents is not required.

ISFSI MALFUNCTION

E-HU1

ECL: Notification of Unusual Event

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.

Basis:

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the sealed loaded storage cask is located in its final overpack. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category A IC AU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

Developer Notes:

The results of the ISFSI Safety Analysis Report (SAR) [per NUREG 1536], or a SAR referenced in the cask Certificate of Compliance and the related NRC Safety Evaluation Report, identify the natural phenomena events and accident conditions that could potentially affect the CONFINEMENT BOUNDARY. This EAL addresses damage that could result from the range of identified natural or man-made events (e.g., a dropped or tipped over cask, EXPLOSION, FIRE, EARTHQUKE, etc.).

The allowable radiation level for a spent fuel cask can be found in the cask's technical specification located in the Certificate of Compliance.

ECL Assignment Attributes: 3.1.1.B

9 FISSION PRODUCT BARRIER ICS/EALS

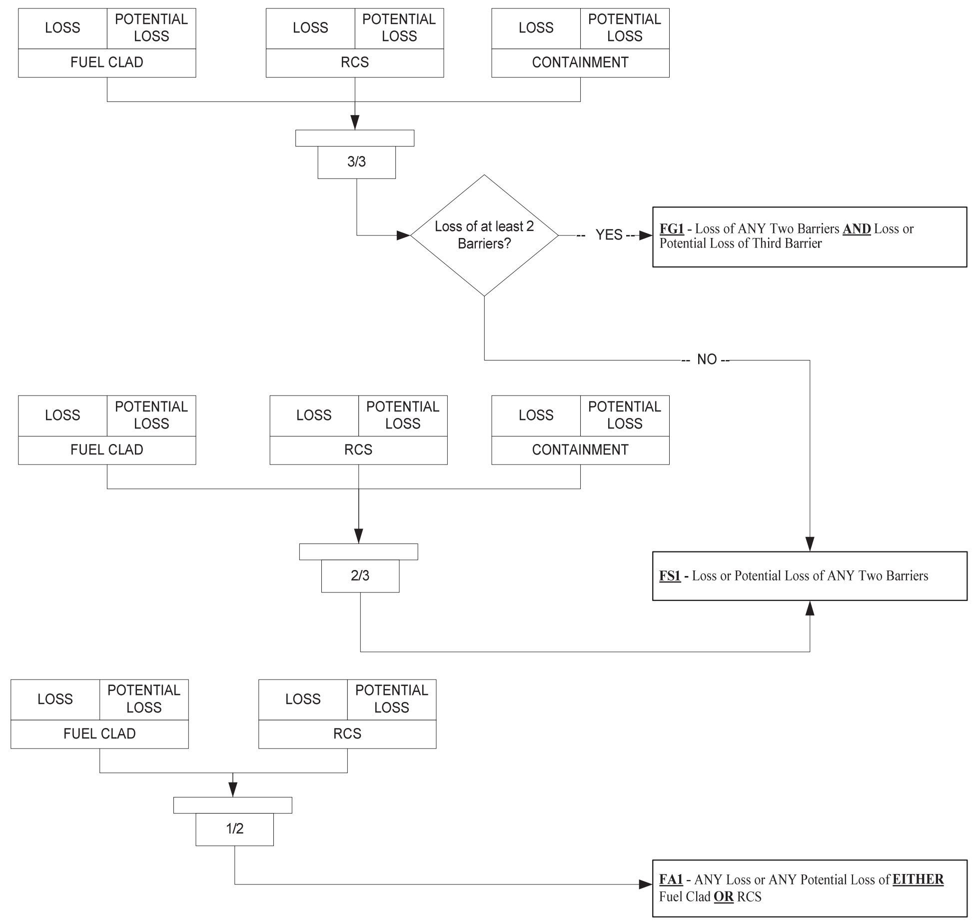
Table 9-F-1: Recognition Category "F" Initiating Condition Matrix

ALERT	
FA1	Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*
SITE AREA EMERGENCY	
FS1	Loss or Potential Loss of any two barriers. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*
GENERAL EMERGENCY	
FG1	Loss of any two barriers and Loss or Potential Loss of the third barrier. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*

See Table 9-F-2 for ESBWR EALs

See Table 9-F-3 for AP1000 EALs

Developer Note: The adjacent logic flow diagram is for use by developers and is not required for site-specific implementation; however, a site-specific scheme must include some type of user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. Such aids are typically comprised of logic flow diagrams, "scoring" criteria or checkbox-type matrices. The user-aid logic must be consistent with that of the adjacent diagram.



Developer Notes

1. The logic used for these initiating conditions reflects the following considerations:

    - The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.

    - Unusual Event ICs associated with fission product barriers are addressed in Recognition Category S.

2. For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC AG1 has been exceeded.

3. The fission product barrier thresholds specified within a scheme are expected to reflect plant-specific design and operating characteristics. This may require that developers create different thresholds than those provided in the generic guidance.

4. Alternative presentation methods for the Recognition Category F ICs and fission product barrier thresholds are acceptable and include flow charts, block diagrams, and checklist-type tables. Developers must ensure that the site-specific method addresses all possible threshold combinations and classification outcomes shown in the ESBWR or AP1000 EAL fission product barrier tables. The NRC staff considers the presentation method of the Recognition Category F information to be an important user aid and may request a change to a particular proposed method if, among other reasons, the change is necessary to promote consistency across the industry.

5. As used in this Recognition Category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside containment, a secondary-side system (i.e., AP1000 steam generator tube leakage), an interfacing system, or outside of containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.

6. At the Site Area Emergency level, classification decision-makers should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

7. The ability to escalate to a higher emergency classification level in response to degrading conditions should be maintained. For example, a steady increase in RCS leakage would represent an increasing risk to public health and safety.

Table 9-F-2: ESBWR EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

FA1 ALERT	FS1 SITE AREA EMERGENCY	FG1 GENERAL EMERGENCY
Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.	Loss or Potential Loss of any two barriers.	Loss of any two barriers and Loss or Potential Loss of the third barrier.

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. RCS Activity		**1. Primary Containment Pressure**		**1. Primary Containment Conditions**	
A. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).	Not Applicable	A. Primary containment pressure greater than (site-specific value) due to RCS leakage.	Not Applicable	A. UNPLANNED rapid drop in primary containment pressure following primary containment pressure rise **OR** B. Primary containment pressure response not consistent with LOCA conditions.	A. Primary containment pressure greater than (site-specific value) **OR** B. (site-specific explosive mixture) exists inside primary containment **OR** C. HCTL exceeded.
2. RPV Water Level		**2. RPV Water Level**		**2. RPV Water Level**	
A. SAG entry required.	A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) or cannot be	A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active	Not Applicable	Not Applicable	A. SAG entry required.

<table>
<tr><th colspan="2">Fuel Clad Barrier</th><th colspan="2">RCS Barrier</th><th colspan="2">Containment Barrier</th></tr>
<tr><th>LOSS</th><th>POTENTIAL LOSS</th><th>LOSS</th><th>POTENTIAL LOSS</th><th>LOSS</th><th>POTENTIAL LOSS</th></tr>
<tr><td></td><td>determined.</td><td>fuel) or cannot be determined.</td><td></td><td></td><td></td></tr>
<tr><td colspan="2">3. Not Applicable</td><td colspan="2">3. RCS Leak Rate</td><td colspan="2">3. Primary Containment Isolation Failure</td></tr>
<tr><td>Not Applicable</td><td>Not Applicable</td><td>A. UNISOLABLE break in ANY of the following: (site-specific systems with potential for high-energy line breaks)
OR
B. Emergency RPV Depressurization.</td><td>A. UNISOLABLE primary system leakage that results in exceeding EITHER of the following:
1. Max Normal Operating Temperature
OR
2. Max Normal Operating Area Radiation Level.</td><td>A. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal
OR
B. Intentional primary containment venting per EOPs
OR
C. UNISOLABLE primary system leakage that results in exceeding EITHER of the following:
1. Max Safe Operating Temperature.
OR
2. Max Safe Operating Area Radiation Level.</td><td>Not Applicable</td></tr>
</table>

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
4. Primary Containment Radiation		**4. Primary Containment Radiation**		**4. Primary Containment Radiation**	
A. Primary containment radiation monitor reading greater than (site-specific value).	Not Applicable	A. Primary containment radiation monitor reading greater than (site-specific value).	Not Applicable	Not Applicable	A. Primary containment radiation monitor reading greater than (site-specific value).
5. Other Indications		**5. Other Indications**		**5. Other Indications**	
A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)
6. Emergency Director Judgment		**6. Emergency Director Judgment**		**6. Emergency Director Judgment**	
A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis Information For
ESBWR EAL Fission Product Barrier Table 9-F-2

ESBWR FUEL CLAD BARRIER THRESHOLDS:

The Fuel Clad barrier consists of the zircalloy fuel bundle tubes that contain the fuel pellets.

1. RCS Activity

Loss 1.A

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

There is no Potential Loss threshold associated with RCS Activity.

Developer Notes:

Threshold values should be determined assuming RCS radioactivity concentration equals 300 μCi/gm dose equivalent I-131. Other site-specific units may be used (e.g., μCi/cc).

Depending upon site-specific capabilities, this threshold may have a sample analysis component and/or a radiation monitor reading component.

Add this paragraph (or similar wording) to the Basis if the threshold includes a sample analysis component, "It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications."

2. RPV Water Level

Loss 2.A

The Loss threshold represents the EOP requirement for SAG entry. The Loss threshold represents the EOP requirement for exit from the symptom-based EOPs and entry into the Severe Accident Guidelines (SAGs). Severe accidents are generally defined to begin with the onset of core damage. The SAGs coordinate control of key plant parameters under severe accident conditions and provide guidance on flooding the primary containment if appropriate to submerge the core and core debris.

SAG entry is required if the core cannot be adequately cooled using all available RPV injection sources:

- In non-ATWS events and RPV water level can be determined, RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level and spray cooling conditions cannot be established, after having emergency depressurized the RPV.

ESBWR FUEL CLAD BARRIER THRESHOLDS:

- In ATWS events and RPV water level can be determined, RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level, after having emergency depressurized the RPV
- If RPV water level *cannot* be determined, it is determined that core damage is occurring while RPV flooding is in progress

Potential Loss 2.A

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold 2.A. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term “cannot be restored and maintained above” means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *cannot* be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA5 or SS5 will dictate the need for emergency classification.

ESBWR FUEL CLAD BARRIER THRESHOLDS:

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

Developer Notes:

<u>Loss 2.A</u>

The phrase, "SAG entry required," should be modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs.

<u>Potential Loss 2.A</u>

The decision that "RPV water level cannot be determined" is directed by guidance given in the RPV water level control sections of the EOPs.

3. Not Applicable (included for numbering consistency between barrier tables)

4. Primary Containment Radiation

<u>Loss 4.A</u>

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold 4.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Radiation.

Developer Notes:

The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS radioactivity concentration equal to 300 μCi/gm dose equivalent I-131, into the primary containment atmosphere.

ESBWR FUEL CLAD BARRIER THRESHOLDS:

5. Other Indications

Loss and/or Potential Loss 5.A

This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Fuel Clad barrier based on plant-specific design characteristics not considered in the generic guidance.

Developer Notes:

Loss and/or Potential Loss 5.A

Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.

Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Developer Notes:

None

ESBWR RCS BARRIER THRESHOLDS:

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

1. Primary Containment Pressure

Loss 1.A

The (site-specific value) primary containment pressure is the drywell high pressure setpoint which indicates a LOCA by automatically initiating ECCS or equivalent makeup system.

There is no Potential Loss threshold associated with Primary Containment Pressure.

Developer Notes:

None

2. RPV Water Level

Loss 2.A

This water level corresponds to the top of active fuel and is used in the EOPs to indicate challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold 2.A. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure and gravity drain injection sources to restore RPV water level or 2) no low pressure or gravity drain RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

ESBWR RCS BARRIER THRESHOLDS:

The term, "cannot be restored and maintained above," means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *cannot* be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation beyond the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA5 or SS5 will dictate the need for emergency classification.

There is no RCS Potential Loss threshold associated with RPV Water Level.

3. RCS Leak Rate

<u>Loss Threshold 3.A</u>

Large high-energy lines that rupture outside primary containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated from the Control Room, the RCS barrier Loss threshold is met.

<u>Loss Threshold 3.B</u>

Emergency RPV Depressurization, either automatically by ADS or manually in accordance with the EOPs is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is manually performed, the plant operators are directed to open safety relief valves (SRVs) and keep them open. Even though the RCS is being vented into the suppression pool, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

<u>Potential Loss Threshold 3.A</u>

Potential loss of RCS based on primary system leakage outside the primary containment is determined from EOP temperature or radiation Max Normal Operating values in areas such as main steam line tunnel, IC, RWCU/SDC, etc., which indicate a direct path from the RCS to areas outside primary containment.

A Max Normal Operating value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

ESBWR RCS BARRIER THRESHOLDS:

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

An UNISOLABLE leak which is indicated by Max Normal Operating values escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold 3.A (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

Developer Notes:

<u>Loss Threshold 3.A</u>

The list of systems included in this threshold should be the high energy lines which, if ruptured and remain unisolated, can rapidly depressurize the RPV. These lines are typically isolated by actuation of the Leak Detection & Isolation System (LD&IS).

Large high-energy line breaks such as Main Steam Line (MSL), Feedwater, Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) or Isolation Condenser (IC) that are UNISOLABLE represent a significant loss of the RCS barrier.

4. Primary Containment Radiation

<u>Loss 4.A</u>

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold 4.A since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with Primary Containment Radiation.

Developer Notes:

The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS activity at Technical Specification allowable limits, into the primary containment atmosphere. Using RCS activity at Technical Specification allowable limits aligns this threshold with IC SU3. Also, RCS activity at this level will typically result in primary containment radiation levels that can be more readily detected by primary containment radiation monitors, and more readily differentiated from those caused by piping or component “shine” sources. If desired, a plant may use a lesser value of RCS activity for determining this value.

ESBWR RCS BARRIER THRESHOLDS:

In some cases, the site-specific physical location and sensitivity of the primary containment radiation monitor(s) may be such that radiation from a cloud of released RCS gases cannot be distinguished from radiation emanating from piping and components containing elevated reactor coolant activity. If so, refer to the Developer Guidance for Loss/Potential Loss 5.A and determine if an alternate indication is available.

5. Other Indications

<u>Loss and/or Potential Loss 5.A</u>

This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the RCS barrier based on plant-specific design characteristics not considered in the generic guidance.

Developer Notes:

<u>Loss and/or Potential Loss 5.A</u>

Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.

Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.

6. Emergency Director Judgment

<u>Loss 6.A</u>

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost.

<u>Potential Loss 6.A</u>

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Developer Notes:

None

ESBWR CONTAINMENT BARRIER THRESHOLDS:

The Primary Containment Barrier includes the drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

1. Primary Containment Conditions

Loss 1.A and 1.B

Rapid UNPLANNED loss of primary containment pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of primary containment integrity. Primary containment pressure should increase as a result of mass and energy release into the primary containment from a LOCA. Thus, primary containment pressure not increasing under these conditions indicates a loss of primary containment integrity.

These thresholds rely on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Potential Loss 1.A

The threshold pressure is the primary containment internal design pressure. Structural acceptance testing demonstrates the capability of the primary containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

Potential Loss 1.B

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the primary containment, loss of the Containment barrier could occur.

Potential Loss 1.C

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,

  OR

ESBWR CONTAINMENT BARRIER THRESHOLDS:

- Suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

Developer Notes:

Potential Loss 1.B

BWR EPGs/SAGs specifically define the limits associated with explosive mixtures in terms of deflagration concentrations of hydrogen and oxygen. Typically the deflagration limits are 6% hydrogen and 5% oxygen in the drywell or suppression chamber.

Potential Loss 1.C

None

2. RPV Water Level

There is no Loss threshold associated with RPV Water Level.

Potential Loss 2.A

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold 2.A. The Potential Loss threshold represents the EOP requirement for exit from the symptom-based EOPs and entry into the Severe Accident Guidelines (SAGs). Severe accidents are generally defined to begin with the onset of core damage. The SAGs coordinate control of key plant parameters under severe accident conditions and provide guidance on flooding the primary containment if appropriate to submerge the core and core debris.

SAG entry is required if the core cannot be adequately cooled using all available RPV injection sources:

- In non-ATWS events and RPV water level can be determined, RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level and spray cooling conditions cannot be established, after having emergency depressurized the RPV
- In ATWS events and RPV water level can be determined, RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level, after having emergency depressurized the RPV
- If RPV water level *cannot* be determined, it is determined that core damage is occurring while RPV flooding is in progress.

ESBWR CONTAINMENT BARRIER THRESHOLDS:

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and increased potential for primary containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

Developer Notes:

The phrase, "SAG entry required," should be modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs.

3. Primary Containment Isolation Failure

These thresholds address incomplete containment isolation that allows an UNISOLABLE direct release to the environment.

Loss 3.A

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the primary containment isolation function.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Following the leakage of RCS mass into primary containment and a rise in primary containment pressure, there may be minor radiological releases associated with allowable primary containment leakage through various penetrations or system components. Minor releases may also occur if a primary containment isolation valve(s) fails to close but the primary containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of primary containment but should be evaluated using the Recognition Category A ICs.

Loss 3.B

EOPs may direct primary containment isolation valve logic(s) to be intentionally bypassed, even if offsite radioactivity release rate limits will be exceeded. Under these conditions with a valid primary containment isolation signal, the containment should also be considered lost if primary containment venting is actually performed.

Intentional venting of primary containment for primary containment pressure or combustible gas control to the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

ESBWR CONTAINMENT BARRIER THRESHOLDS:

Loss 3.C

The Max Safe Operating Temperature and the Max Safe Operating Radiation Level are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

The temperatures and radiation levels should be confirmed to be caused by RCS leakage from a primary system. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

In combination with RCS potential loss 3.A this threshold would result in a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Isolation Failure.

Developer Notes:

Loss 3.B

Consideration may be given to specifying the specific procedural step within the Primary Containment Control EOP that defines intentional venting of the Primary Containment regardless of offsite radioactivity release rate.

4. Primary Containment Radiation

There is no Loss threshold associated with Primary Containment Radiation.

Potential Loss 4.A

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.

ESBWR CONTAINMENT BARRIER THRESHOLDS:

Developer Notes:

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, provides the basis for using the 20% fuel cladding failure value. Unless there is a site-specific analysis justifying a different value, the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad failure into the primary containment atmosphere.

5. Other Indications

<u>Loss and/or Potential Loss 5.A</u>

This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Containment barrier based on plant-specific design characteristics not considered in the generic guidance.

Developer Notes:

<u>Loss and/or Potential Loss 5.A</u>

Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.

Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.

6. Emergency Director Judgment

<u>Loss 6.A</u>

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost.

<u>Potential Loss 6.A</u>

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Developer Notes:

None

Table 9-F-3: AP1000 EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

FA1 ALERT	FS1 SITE AREA EMERGENCY	FG1 GENERAL EMERGENCY
Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.	Loss or Potential Loss of any two barriers.	Loss of any two barriers and Loss or Potential Loss of the third barrier.

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. RCS or SG Tube Leakage		**1. RCS or SG Tube Leakage**		**1. RCS or SG Tube Leakage**	
Not Applicable	A. (site-specific RCS/reactor vessel level) less than (site-specific level).	A. An automatic or manual safeguards actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE. **OR** B. Automatic or manual actuation of ADS	A. Operation of a second CVS makeup pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. Inadequate Heat Removal		**2. Inadequate Heat Removal**		**2. Inadequate Heat Removal**	
A. Core exit thermocouple readings greater than (site-specific temperature value).	A. Core exit thermocouple readings greater than (site-specific temperature value). **OR** B. Inadequate RCS heat removal capability via steam generators or PRHR HX as indicated by (site-specific indications).	Not Applicable	A. Inadequate RCS heat removal capability via steam generators or PRHR HX as indicated by (site-specific indications).	Not Applicable	A. 1. (Site-specific criteria for entry into core cooling restoration procedure) **AND** 2. Restoration procedure not effective within 15 minutes.
3. RCS Activity / Containment Radiation		**3. RCS Activity / Containment Radiation**		**3. RCS Activity / Containment Radiation**	
A. Containment radiation monitor reading greater than (site-specific value). **OR** B. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).	Not Applicable	A. Containment radiation monitor reading greater than (site-specific value).	Not Applicable	Not Applicable	A. Containment radiation monitor reading greater than (site-specific value).

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
4. Containment Integrity or Bypass		**4. Containment Integrity or Bypass**		**4. Containment Integrity or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Containment isolation is required **AND** **EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of unisolable RCS leakage outside of containment.	A. Containment pressure greater than (site-specific value) **OR** B. Explosive mixture exists inside containment **OR** C. 1. Containment pressure greater than (site-specific pressure setpoint) **AND** 2. PCS does not actuate per design for 15 minutes or longer.
5. Other		**5. Other**		**5. Other**	
A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)
6. Emergency Director Judgment		**6. Emergency Director Judgment**		**6. Emergency Director Judgment**	
A. **ANY** condition in the opinion of	A. **ANY** condition in the opinion of the	A. **ANY** condition in the opinion of the	A. **ANY** condition in the opinion of the	A. **ANY** condition in the opinion of the	A. **ANY** condition in the opinion of the

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
the Emergency Director that indicates Loss of the Fuel Clad Barrier.	Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Emergency Director that indicates Loss of the RCS Barrier.	Emergency Director that indicates Potential Loss of the RCS Barrier.	Emergency Director that indicates Loss of the Containment Barrier.	Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis Information For

AP1000 EAL Fission Product Barrier Table 9-F-3

Developer Notes:

Threshold Parameters and Values

Each PWR owner's group has developed a methodology for guiding the development and implementation of EOPs (i.e., assessing plant parameters, and determining and prioritizing operator actions). Many of the thresholds contained in the AP1000 EAL Fission Product Barrier Table reflect conditions that are addressed in EOPs (e.g., a loss of heat removal capability by the steam generators and PRHR). When developing a site-specific threshold, developers should use the parameters and values specified within their EOPs that align with the condition described by the generic threshold and basis, and related developer notes. This approach will ensure consistency between the site-specific EOPs and emergency classification scheme, and thus facilitate more timely and accurate classification assessments.

In support of EOP development and implementation, the Westinghouse Owners Group (WOG) developed a defined set of Critical Safety Functions as part of their Emergency Response Guidelines. The WOG approach structures EOPs to maintain and/or restore these Critical Safety Functions, and to do so in a prioritized and systematic manner. The WOG Critical Safety Functions are presented below.

- Subcriticality
- Core Cooling
- Heat Sink
- RCS Integrity
- Containment
- RCS Inventory

The WOG ERGs provide a methodology for monitoring the status of the Critical Safety Functions and classifying the significance of a challenge to a function; this methodology is referred to as the Critical Safety Function Status Trees (CSFSTs). For AP1000 plants, the guidance in NEI 07-01 allows for use of certain CSFST assessment results as EALs and fission product barrier loss/potential loss thresholds. In this manner, an emergency classification assessment may flow directly from a CSFST assessment.

It is important to understand that the CSFSTs are evaluated using plant parameters, and that they are simply a vendor-specific method for collectively evaluating a set of parameters for purposes of driving emergency operating procedure usage. For the emergency conditions of interest, the generic thresholds within the AP1000 EAL Fission Product Barrier Table specify the plant parameters that define a potential loss or loss of a fission product barrier; however, as described

in the associated Developer Notes, a CSFST terminus may be used as well. For this reason, inclusion of the CSFST-related thresholds would be redundant to the parameter-based thresholds.

Sites may, at their discretion, include the CSFST-based loss and potential loss thresholds as described in the Developer Notes. Developers at these sites should consult with their classification decision-makers to determine if inclusion would assist with timely and accurate emergency classification. This decision should consider the effects of any site-specific changes to the generic WOG CSFST evaluation logic and setpoints, as well as those arising from user rules applicable to emergency operating procedures (e.g., exceptions to procedure entry or transition due to specific accident conditions or loss of a support system).

The CSFST thresholds may be addressed in one of 3 ways:

1) Not incorporated; thresholds will use parameters and values as discussed in the Developer Notes.

2) Incorporated along with parameter and value thresholds (e.g., a fuel clad loss would have 2 thresholds such as “CETs > 1200^{o}F” and “Core Cooling Red entry conditions met”.

3) Used in lieu of parameters and values for all thresholds.

With one exception, if a decision is made to include the CSFST-based thresholds, then all such allowed thresholds must be used in the table (e.g., it is not permissible to use only the C Orange terminus as a potential loss of the fuel clad barrier threshold and disregard all other CSFST-based thresholds). The one exception is the RCS Integrity (P) CSFST. Because of the complexity of the P Red decision-point that relies on an assessment a pressure-temperature curve, a P Red condition may be used as an RCS potential loss threshold without the need to incorporate the other CSFST-based thresholds.

AP1000 FUEL CLAD BARRIER THRESHOLDS:

The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

1. RCS or SG Tube Leakage

There is no Loss threshold associated with RCS or SG Tube Leakage.

Potential Loss 1.A

This reading indicates a reduction in reactor vessel water level that threatens the onset of heat-induced cladding damage.

Developer Notes:

Potential Loss 1.A

Enter the site-specific RCS/reactor vessel water level value(s) used by EOPs to identify a degraded core cooling condition (e.g., requires prompt restoration action). The minimum observable RCS hot leg level may also be used.

2. Inadequate Heat Removal

Loss 2.A

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

Potential Loss 2.A

This reading indicates temperatures within the core are sufficient to allow the onset of heat-induced cladding damage.

Potential Loss 2.B

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators or PRHR heat exchangers (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted. This threshold is also not applicable to conditions where heat sink is not required (i.e. steam generator pressures are greater than RCS pressure).

Meeting this threshold results in a Site Area Emergency because this threshold is identical to RCS Barrier Potential Loss threshold 2.A; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

AP1000 FUEL CLAD BARRIER THRESHOLDS:

Developer Notes:

Some site-specific EOPs and/or EOP user guidelines may establish decision-making criteria concerning the number or other attributes of thermocouple readings necessary to drive actions (e.g., 5 CETs reading greater than 1,200^{o}F is required before transitioning to an inadequate core cooling procedure). To maintain consistency with EOPs, these decision-making criteria may be used in the core exit thermocouple reading thresholds.

Loss 2.A

Enter a site-specific temperature value that corresponds to significant in-core superheating of reactor coolant. 1,200^{o}F may also be used.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Red Path. Alternatively, as a loss indication, developers may consider including a threshold the same as, or similar to, “Core Cooling Red entry conditions met” in accordance with the guidance at the front of this section.

Potential Loss 2.A

Enter a site-specific temperature value that corresponds to core conditions at the onset of heat-induced cladding damage (e.g., the temperature allowing for the formation of superheated steam assuming that the RCS is intact). 700ºF may also be used.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Orange Path. Alternatively, developers may consider including a threshold the same as, or similar to, “Core Cooling Orange entry conditions met” in accordance with the guidance at the front of this section.

Potential Loss 2.B

Enter the site-specific parameters and values that define an extreme challenge to the ability to remove heat from the RCS via the steam generators and PRHR. These will typically be parameters and values that would require operators to take prompt action to address this condition.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Heat Sink Red Path. Alternatively, developers may consider including a threshold the same as, or similar to, “Heat Sink Red entry conditions met” in accordance with the guidance at the front of this section.

AP1000 FUEL CLAD BARRIER THRESHOLDS:

3. RCS Activity / Containment Radiation

<u>Loss 3.A</u>

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold 3.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

<u>Loss 3.B</u>

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

Developer Notes:

<u>Loss 3.A</u>

The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS radioactivity concentration equal to 300 μCi/gm dose equivalent I-131, into the containment atmosphere.

Alternatively a Containment High Range Radiation Monitor alarm setpoint can be specified provided the alarm setpoint is indicative of 2 to 5% fuel clad failure released into containment.

AP1000 FUEL CLAD BARRIER THRESHOLDS:

Loss 3.B

Threshold values should be determined assuming RCS radioactivity concentration equals 300 μCi/gm dose equivalent I-131. Other site-specific units may be used (e.g., μCi/cc).

Depending upon site-specific capabilities, this threshold may have a sample analysis component and/or a radiation monitor reading component.

Add this paragraph (or similar wording) to the Basis if the threshold includes a sample analysis component, "It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications."

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

5. Other Indications

Loss and/or Potential Loss 5.A

This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Fuel Clad barrier based on plant-specific design characteristics not considered in the generic guidance.

Developer Notes:

Loss and/or Potential Loss 5.A

Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.

Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

AP1000 FUEL CLAD BARRIER THRESHOLDS:

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Developer Notes:

None

AP1000 RCS BARRIER THRESHOLDS:

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.

1. RCS or SG Tube Leakage

Loss 1.A

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual safeguards actuation. This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

Loss 1.B

The Automatic Depressurization System (ADS) valves are part of the RCS and interface with the passive core cooling system (PXS). Opening the ADS valves is required for the PXS to function. Manual or automatic actuation of the ADS valves in response to a valid initiation setpoint initiates an RCS leak meeting the RCS Loss criteria. However, inadvertent isolable opening or momentary jogging of ADS valves for pressure control does not constitute a loss of RCS.

Potential Loss 1.A

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a single CVS makeup pump, but safeguards actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a second CVS makeup pump be placed in service to restore and maintain pressurizer level. Nominal design flow rate of a single CVS makeup pump is (site-specific gpm).

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

AP1000 RCS BARRIER THRESHOLDS:

Potential Loss 1.B

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

Developer Notes:

Loss 1.A

Actuation of safeguards may also be referred to as Safety Injection (SI) actuation or other appropriate site-specific term.

Potential Loss 1.A

Developers may use an RCS leak rate value of 135 gpm (nominal design flow rate of a single CVS makeup pump), or an appropriate site-specific value, as an alternate Potential Loss threshold. If used, the threshold wording should reflect that the determination of the leak rate value excludes normal reductions in RCS inventory (e.g., by CVS letdown).

Potential Loss 1.B

Enter the site-specific indications that define an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized). These will typically be parameters and values that would require operators to take prompt action to address a pressurized thermal shock condition. Developers should also determine if the threshold needs to reflect any dependencies used as EOP transition/entry decision points or condition validation criteria (e.g., an EOP used to respond to an excessive RCS cooldown may not be entered or immediately exited if RCS pressure is below a certain value).

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the RCS Integrity Red Path. Because of the complexity of certain decision-points within the Red Path of this CSFST, developers at these plants may elect to not include the specific parameters and values. Alternatively, as a potential loss indication, developers may consider including a threshold the same as, or similar to, "RCS Integrity Red entry conditions met" in accordance with the guidance at the front of this section. As noted above, developers should ensure that the threshold wording reflects any EOP transition/entry decision points or condition validation criteria. For example, a threshold might read "RCS Integrity (P) Red entry conditions met with RCS pressure > 300 psig."

AP1000 RCS BARRIER THRESHOLDS:

2. Inadequate Heat Removal

There is no Loss threshold associated with Inadequate Heat Removal.

Potential Loss 2.A

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators or PRHR (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted. This threshold is also not applicable to conditions where heat sink is not required (i.e. steam generator pressures are greater than RCS pressure).

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold 2.B; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

Developer Notes:

Potential Loss 2.A

Enter the site-specific parameters and values that define an extreme challenge to the ability to remove heat from the RCS via the steam generators or PRHR. These will typically be parameters and values that would require operators to take prompt action to address this condition.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Heat Sink Red Path. Alternatively, developers may consider including a threshold the same as, or similar to, "Heat Sink Red entry conditions met" in accordance with the guidance at the front of this section.

3. RCS Activity / Containment Radiation

Loss 3.A

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold 3.A since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

AP1000 RCS BARRIER THRESHOLDS:

Developer Notes:

Loss 3.A

The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS activity at the transient iodine spike Technical Specification allowable limit (typically > 60 µCi/gm dose equivalent I-131), into the containment atmosphere. RCS activity at this level will typically result in containment radiation levels that can be more readily detected by containment radiation monitors, and more readily differentiated from those caused by piping or component "shine" sources. If desired, a plant may use a lesser value of RCS activity for determining this value.

In some cases, the site-specific physical location and sensitivity of the containment radiation monitor(s) may be such that radiation from a cloud of released RCS gases cannot be distinguished from radiation emanating from piping and components containing elevated reactor coolant activity. If so, refer to the Developer Notes for Loss/Potential Loss 5.A and determine if an alternate indication is available.

Alternatively a Containment High Range Radiation Monitor alarm setpoint can be specified provided the alarm setpoint is indicative of RCS activity at the Technical Specification limit released into containment.

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

5. Other Indications

Loss and/or Potential Loss 5.A

This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the RCS barrier based on plant-specific design characteristics not considered in the generic guidance.

Developer Notes:

Loss and/or Potential Loss 5.A

Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.

Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.

AP1000 RCS BARRIER THRESHOLDS:

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Developer Notes:

None

AP1000 CONTAINMENT BARRIER THRESHOLDS:

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

1. RCS or SG Tube Leakage

Loss 1.A

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss 1.A and Loss 1.A, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 07-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably [*part of the FAULTED definition*] and the faulted steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU3 for the fuel clad barrier (i.e., RCS activity values) and IC SU4 for the RCS barrier (i.e., RCS leak rate values).

Steam releases associated with the expected operation of a SG power operated relief valve (PORV) or Main Steam Safety Valve (MSSV) do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open MSSV) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category A ICs.

The emergency classification levels resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

AP1000 CONTAINMENT BARRIER THRESHOLDS:

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to RCS leakage value specified in IC SU4	No classification	No classification
Greater than RCS leakage value specified in IC SU4	Unusual Event per SU4	Unusual Event per SU4
Requires operation of a second CVS makeup pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1	Alert per FA1
Requires an automatic or manual safeguards actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1	Alert per FA1

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

Developer Notes:

<u>Loss 1.A</u>

Developers may wish to consider incorporating the above table into user aids (e.g., a wallboard) or other locations within their basis document.

2. Inadequate Heat Removal

There is no Loss threshold associated with Inadequate Heat Removal.

<u>Potential Loss 2.A</u>

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

AP1000 CONTAINMENT BARRIER THRESHOLDS:

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events, particularly through use of the In-Vessel Retention System. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

Developer Notes:

Some site-specific EOPs and/or EOP user guidelines may establish decision-making criteria concerning the number or other attributes of thermocouple readings necessary to drive actions (e.g., 5 CETs reading greater than 1,200°F is required before transitioning to an inadequate core cooling procedure). To maintain consistency with EOPs, these decision-making criteria may be used in the core exit thermocouple reading thresholds.

<u>Potential Loss 2.A.1</u>

Enter site-specific criteria requiring entry into a core cooling restoration procedure or prompt implementation of core cooling restoration actions. A reading of 1,200°F on the CETs may also be used.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Red Path. Alternatively, developers may consider including a threshold the same as, or similar to, "Core Cooling Red entry conditions met for 15 minutes or longer" in accordance with the guidance at the front of this section.

AP1000 CONTAINMENT BARRIER THRESHOLDS:

3. RCS Activity / Containment Radiation

There is no Loss threshold associated with RCS Activity / Containment Radiation.

<u>Potential Loss 3.A</u>

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.

Developer Notes:

<u>Potential Loss 3.A</u>

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, provides the basis for using the 20% fuel cladding failure value. Unless there is a site-specific analysis justifying a different value, the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad failure into the containment atmosphere.

Alternatively a Containment High Range Radiation Monitor alarm setpoint can be specified provided the alarm setpoint is indicative of 20% fuel clad failure released into containment.

4. Containment Integrity or Bypass

<u>Loss 4.A</u>

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both thresholds 4.A.1 and 4.A.2.

AP1000 CONTAINMENT BARRIER THRESHOLDS:

4.A.1 – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Director will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 9-F-4. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category A ICs.

4.A.2 – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure not attributable to PCS actuation.

Refer to the top piping run of Figure 9-F-4. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

AP1000 CONTAINMENT BARRIER THRESHOLDS:

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 9-F-4. In this simplified example, leakage in an RCP cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump or system piping developed a leak that allowed steam/water to enter the Auxiliary Building, then threshold 4.B would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to a closed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category A ICs.

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold 1.A.

<u>Loss 4.B</u>

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure 9-F-4. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold 1.A to be met.

AP1000 CONTAINMENT BARRIER THRESHOLDS:

Potential Loss 4.A

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

Potential Loss 4.B

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

Potential Loss 4.C

This threshold describes a condition where containment pressure is greater than the setpoint at which the containment energy (heat) removal system, Passive Containment Cooling System (PCS) is designed to automatically actuate, but does not actuate and perform its safety function per design. The 15-minute criterion is included to allow operators time to manually align PCS components that may not have automatically actuated, if possible. This threshold represents a potential loss of containment in that the containment heat removal/depressurization systems (not including containment venting strategies) are either lost or performing in a degraded manner.

Developer Notes:

Loss 4.A.1

Developers may include a list of site-specific radiation monitors to better define this threshold. Expected monitor alarms or readings may also be included.

Potential Loss 4.A

The site-specific pressure is the containment design pressure.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, the pressure value in Potential Loss 4.A is that used for the Containment Red Path. If the Containment CSFST contains more than one Red Path due to other dependencies (e.g., status of containment isolation), enter the highest containment pressure value shown on the tree. This is typically the containment design pressure.

AP1000 CONTAINMENT BARRIER THRESHOLDS:

Potential Loss 4.B

Developers may enter the minimum containment atmospheric hydrogen concentration necessary to support a hydrogen burn (i.e., the lower deflagration limit). A concurrent containment oxygen concentration may be included if the plant has this indication available in the Control Room.

Potential Loss 4.C

Enter the site-specific pressure setpoint value that actuates PCS. If desired, specific condition indications such as parameter values can also be entered that indicate a failure of PCS to actuate per design.

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As a potential loss indication, developers should consider including a threshold the same as, or similar to, “Containment Red entry conditions met” in accordance with the guidance at the front of this section.

5. Other Indications

Loss and/or Potential Loss 5.A

This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Containment barrier based on plant-specific design characteristics not considered in the generic guidance.

Developer Notes:

Loss and/or Potential Loss 5.A

If site emergency operating procedures provide for venting of the containment as a means of preventing catastrophic failure, a Loss threshold should be included for the containment barrier. This threshold would be met as soon as such venting is IMMINENT. Containment venting as part of recovery actions is classified in accordance with the radiological effluent ICs.

Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Design Control Document, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.

Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.

AP1000 CONTAINMENT BARRIER THRESHOLDS:

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

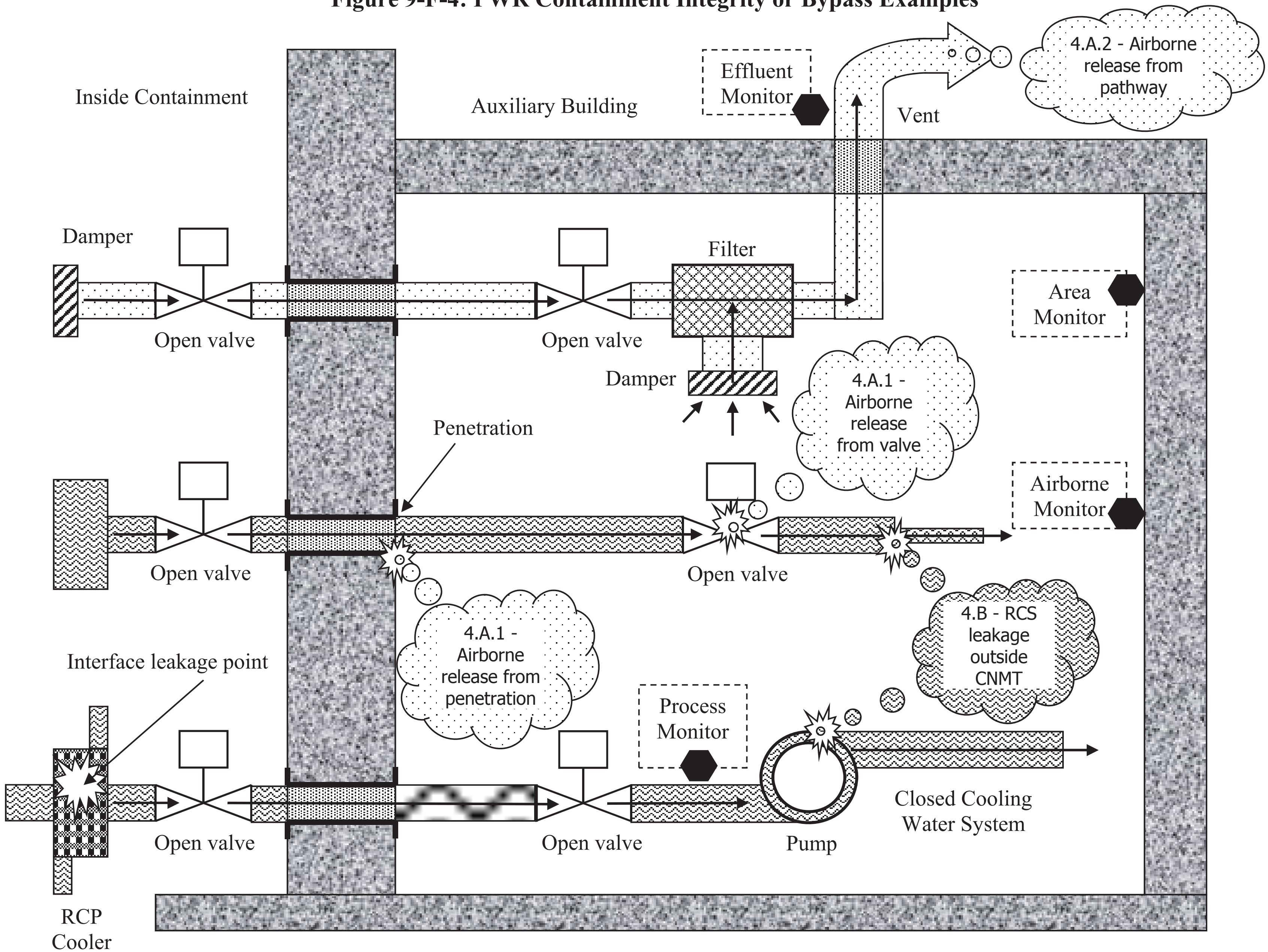
Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Developer Notes:

None

Figure 9-F-4: PWR Containment Integrity or Bypass Examples



10 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICs/EALs

Table H-1: Recognition Category "H" Initiating Condition Matrix

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
HU1 Confirmed SECURITY CONDITION or threat. *Op. Modes: All*	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. *Op. Modes: All*	**HS1** HOSTILE ACTION within the PROTECTED AREA. *Op. Modes: All*	**HG1** HOSTILE ACTION resulting in loss of physical control of the facility. *Op. Modes: All*
HU2 Seismic event greater than (site-specific OBE equivalent) levels. *Op. Modes: All*			
HU3 Hazardous event. *Op. Modes: All*			
HU4 FIRE potentially degrading the level of safety of the plant. *Op. Modes: All*			
	HA5 Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown. *Op. Modes: All*		
	HA6 Control Room evacuation resulting in transfer of plant control to alternate locations. *Op. Modes: All*	**HS6** Inability to control a key safety function from outside the Control Room. *Op. Modes: All*	
HU7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE. *Op. Modes: All*	**HA7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert. *Op. Modes: All*	**HS7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency. *Op. Modes: All*	**HG7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency. *Op. Modes: All*

Table intended for use by EAL developers. Inclusion in licensee documents is not required.

HU1

ECL: Notification of Unusual Event

Initiating Condition: Confirmed SECURITY CONDITION or threat

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2 or 3)

(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).

(2) Notification of a credible security threat directed at the site.

(3) A validated notification from the NRC providing information of an aircraft threat.

Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and OROs.

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

EAL #1 references (site-specific security shift supervision) because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.390 information.

EAL #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with (site-specific procedure).

EAL #3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with (site-specific procedure).

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

such as the Security Plan.

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

Developer Notes:

The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.

The (site-specific procedure) is the procedure(s) used by Control Room and/or Security personnel to determine if a security threat is credible, and to validate receipt of aircraft threat information.

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

With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."

ECL Assignment Attributes: 3.1.1.A

HU2

ECL: Notification of Unusual Event

Initiating Condition: Seismic event greater than (site-specific OBE equivalent) levels

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Seismic event greater than (site-specific equivalent of the Operating Basis Earthquake) as indicated by:

(site-specific indication that a seismic event met or exceeded the site-specific equivalent of the OBE limits)

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than the equivalent of an Operating Basis Earthquake (OBE)[6]. An earthquake greater than the equivalent of an OBE but less than a Safe Shutdown Earthquake (SSE)[7] should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., typical lateral accelerations are in excess of 0.10 - 0.15g). The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

Developer Notes:

The Operating Basis Earthquake has been eliminated as a design requirement for both the AP1000 and ESBWR. However, for both the AP1000 and ESBWR, the OBE equivalent is defined to be 1/3 the SSE (0.30g) ground acceleration, or 0.10g in either the horizontal or vertical plane. The "site-specific indication that a seismic event met or exceeded the equivalent of the OBE limits" should be based on the indications, alarms and displays of site-specific seismic monitoring equipment.

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[6] An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.
[7] An SSE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

Indications described in the EAL should be limited to those that are immediately available to Control Room personnel and which can be readily assessed. Indications available outside the Control Room and/or which require lengthy times to assess (e.g., processing of scratch plates or recorded data) should not be used. The goal is to specify indications that can be assessed within 15-minutes of the actual or suspected seismic event.

For sites that do not have readily assessable indications for the OBE equivalent within the Control Room, developers should use the following alternate EAL (or similar wording).

(1) a. Control Room personnel feel an actual or potential seismic event.

AND

b. The occurrence of a seismic event is confirmed in manner deemed appropriate by the Shift Manager or Emergency Director.

The EAL 1.b statement is included to ensure that a declaration does not result from felt vibrations caused by a non-seismic source (e.g., a dropped heavy load). The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration. It is recognized that this alternate EAL wording may cause a site to declare an Unusual Event while another site, similarly affected but with readily assessable OBE indications in the Control Room, may not.

The above alternate wording may also be used to develop a compensatory EAL for use during periods when a seismic monitoring system capable of detecting an OBE equivalent magnitude seismic event is out-of-service for maintenance or repair.

ECL Assignment Attributes: 3.1.1.A

HU3

ECL: Notification of Unusual Event

Initiating Condition: Hazardous event

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2 or 3 or 4 or 5)

Note: EAL #3 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

(1) A tornado strike within the PROTECTED AREA.

(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

(5) (Site-specific list of natural or technological hazard events)

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL #1 addresses a tornado striking (touching down) within the Protected Area.

EAL #2 addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

EAL #3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

EAL #5 addresses (site-specific description).

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Developer Notes:

The "Site-specific list of natural or technological hazard events" should include other events that may be a precursor to a more significant event or condition, and that are appropriate to the site location and characteristics.

Notwithstanding the events specifically included as EALs above, a "Site-specific list of natural or technological hazard events" need not include short-lived events for which the extent of the damage and the resulting consequences can be determined within a relatively short time frame. In these cases, a damage assessment can be performed soon after the event, and the plant staff will be able to identify potential or actual impacts to plant systems and structures. This will enable prompt definition and implementation of compensatory or corrective measures with no appreciable increase in risk to the public.

To the extent that a short-lived event does cause immediate and significant damage to plant systems and structures, it will be classifiable under the Recognition Category F, S and C ICs and EALs. Events of lesser impact would be expected to cause only small and localized damage. The consequences from these types of events are adequately assessed and addressed in accordance with Technical Specifications. In addition, the occurrence or effects of the event may be reportable under the requirements of 10 CFR 50.72.

ECL Assignment Attributes: 3.1.1.A and 3.1.1.C

HU4

ECL: Notification of Unusual Event

Initiating Condition: FIRE potentially degrading the level of safety of the plant

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2 or 3 or 4)

Note: The Emergency Director should declare the Unusual Event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

(1) a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications:

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

b. The FIRE is located within **ANY** of the following plant rooms or areas:

(site-specific list of plant rooms or areas)

(2) a. Receipt of a single fire alarm (i.e., no other indications of a FIRE).

AND

b. The FIRE is located within **ANY** of the following plant rooms or areas:

(site-specific list of plant rooms or areas)

AND

c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

(3) A FIRE within the plant *or ISFSI* [*for plants with an ISFSI outside the plant Protected Area*] PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

(4) A FIRE within the plant *or ISFSI* [*for plants with an ISFSI outside the plant Protected Area*] PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL #1

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

EAL #2

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

EAL #3

In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. *This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.* [*Sentence for plants with an ISFSI outside the plant Protected Area*]

EAL #4

If a FIRE within the plant *or ISFSI* [*for plants with an ISFSI outside the plant Protected Area*] PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency

(e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

Developer Notes:

The “site-specific list of plant rooms or areas” should specify those rooms or areas that contain SAFETY SYSTEM equipment.

As noted in the EALs and Basis section, include the term ISFSI if the site has an ISFSI outside the plant Protected Area.

ECL Assignment Attributes: 3.1.1.A

HU7

ECL: Notification of Unusual Event

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE

Operating Mode Applicability: All

Example Emergency Action Levels:

(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.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a NOUE.

HA1

ECL: Alert

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

Operating Mode Applicability: All

Example Emergency Action Levels: (1 or 2)

(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).

(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

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

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

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

EAL #1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA.

EAL #2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened

state of readiness. This EAL is met when the threat-related information has been validated in accordance with (site-specific procedure).

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

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

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

Escalation of the emergency classification level would be via IC HS1.

Developer Notes:

The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.

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

With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as “Security event #2, #5 or #9 is reported by the (site-specific security shift supervision).”

See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the OWNER CONTROLLED AREA.

ECL Assignment Attributes: 3.1.2.D

HA5

ECL: Alert

Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown

Operating Mode Applicability: All

Example Emergency Action Levels:

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

(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas:

(site-specific list of plant rooms or areas with entry-related mode applicability identified)

AND

b. Entry into the room or area is prohibited or impeded.

Basis:

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and

the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.

- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area.

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

Developer Notes:

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

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

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

ECL Assignment Attributes: 3.1.2.B

HA6

ECL: Alert

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS6.

Developer Notes:

The "site-specific remote shutdown panels and local control stations" are the panels and control stations referenced in plant procedures used to cool down and shut down the plant from a location(s) outside the Control Room.

For the AP1000 the "site-specific remote shutdown panels and local control stations" are located in the Remote Shutdown Room.

ECL Assignment Attributes: 3.1.2.B

HA7

ECL: Alert

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert

Operating Mode Applicability: All

Example Emergency Action Levels:

(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.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for an Alert.

HS1

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).

Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

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

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

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

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

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

Developer Notes:

The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.

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

With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."

See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the PROTECTED AREA.

ECL Assignment Attributes: 3.1.3.D

HS6

ECL: Site Area Emergency

Initiating Condition: Inability to control a key safety function from outside the Control Room

Operating Mode Applicability: All

Example Emergency Action Levels:

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that (site-specific number of minutes) has been exceeded, or will likely be exceeded.

(1) a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).

AND

b. Control of **ANY** of the following key safety functions is not reestablished within (site-specific number of minutes).

- Reactivity control
- Core cooling [*AP1000*] / RPV water level [*ESBWR*]
- RCS heat removal

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within (the site-specific time for transfer) minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

Escalation of the emergency classification level would be via IC FG1 or CG1.

Developer Notes:

The "site-specific remote shutdown panels and local control stations" are the panels and control stations referenced in plant procedures used to cool down and shut down the plant from a location(s) outside the Control Room.

The "site-specific number of minutes" is the time in which plant control must be (or is expected to be) reestablished at an alternate location as described in the site-specific fire response analyses. Absent a basis in the site-specific analyses, 60 minutes should be used. Another time period may be used with appropriate basis/justification.

Passive reactors are designed such that no operator action is required for 72 hours following a Control Room evacuation. The site-specific allowed time should provide sufficient time for Control Room evacuation, assessment of conditions and establishing plant control from the Remote Shutdown Panels.

ECL Assignment Attributes: 3.1.3.B

HS7

ECL: Site Area Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a Site Area Emergency.

HG1

ECL: General Emergency

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).

AND

b. **EITHER** of the following has occurred:

1. **ANY** of the following safety functions cannot be restored or maintained.

   - Reactivity control
   - Core cooling [*AP1000*] / RPV water level [*ESBWR*]
   - RCS heat removal

   OR

2. Damage to spent fuel has occurred or is IMMINENT.

Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained or, 3) physical damage caused by terrorist attack.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

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

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

such as the Security Plan.

Developer Notes:

The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.

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

With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."

See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the PROTECTED AREA.

ECL Assignment Attributes: 3.1.4.D

HG7

ECL: General Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency

Operating Mode Applicability: All

Example Emergency Action Levels:

(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 offsite for more than the immediate site area.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a General Emergency.

11 SYSTEM MALFUNCTION ICS/EALS

Table S-1: Recognition Category "S" Initiating Condition Matrix

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU1 Loss of all ability to charge at least one (site-specific Class 1E battery) for 30 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*	**SA1** Loss of all ability to charge at least one (site-specific Class 1E battery) for 60 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*	**SS1** Loss of all required (site-specific Class 1E DC power) or (site-specific Class 1E UPS bus power) for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*	
	SA2 Partial loss of monitoring or control functions for 15 minutes or longer *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*	**SS2** Complete loss of monitoring or control functions for 15 minutes or longer *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*	
SU3 Reactor coolant activity greater than Technical Specification allowable limits. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*			
SU4 RCS leakage for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*			

Table intended for use by EAL developers. Inclusion in licensee documents is not required.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU5 Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor. *Op. Modes: Power Operation*	**SA5** Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor. *Op. Modes: Power Operation*	**SS5** Inability to shutdown the reactor causing a challenge to (core cooling [*PWR*] / RPV water level [*BWR*]) or RCS heat removal. *Op. Modes: Power Operation*	
SU6 Loss of all onsite or offsite communications capabilities. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*			
SU7 Failure to isolate containment or loss of containment pressure control. [*AP1000*] *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*			
	SA8 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. *Op. Modes: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown*		

Table intended for use by EAL developers. Inclusion in licensee documents is not required.

SU1

ECL: Notification of Unusual Event

Initiating Condition: Loss of all ability to charge at least one (site-specific Class 1E DC battery) for 30 minutes or longer

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

(1) Loss of all ability to charge at least one (site-specific Class 1E 24-hr. DC battery) for 30 minutes or longer

Basis:

The Class 1E 24-hr. (safety-related) DC system provides electrical power for safety-related and vital control and monitoring instrumentation loads. The offsite AC power system normally supplies power for the unit in cold shutdown, refueling, and defueled conditions. Both the normal offsite and standby onsite AC power systems are non-Class 1E with no Technical Specification requirements. All safety-related functions associated with the unit in cold shutdown and refueling are provided by the safety-related onsite Class 1E DC power systems including the 120V Vital AC power system supplied from the batteries powering the safety-related DC buses through inverters. The Passive ALWRs also have standby diesel generators that are not safety-related. Storage batteries are the safety-related power source for Class 1E electric power. However, this non-Class 1E AC power is required to charge the safety-related DC batteries and maintain long-term safety-related DC bus voltage.

Loss of Class 1E 24-hr. DC power potentially compromises all safety-related plant systems requiring electric power. The event can be classified as an Unusual Event, because the passive design affords additional and redundant means to remove heat passively or restore power to active components.

30 minutes was chosen to allow sufficient time for plant personnel to attempt to establish a viable offsite or diesel generator AC power supply to the site-specific AC power buses required to charge one or more Class 1E 24-hr. DC batteries.

Escalation of the emergency classification level would be via IC SA1 if at least one Class 1E battery cannot be charged within 60 min.

Developer Notes:

Buses ECS-ES-1 and 2 for AP1000 or the PIP A3 and B3 buses for ESBWR are the source of non-Class 1E AC power to the Class 1E DC battery chargers.

ECL Assignment Attributes: 3.1.1.A

SU3

ECL: Notification of Unusual Event

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

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels: (1 or 2)

(1) (Site-specific radiation monitor) reading greater than (site-specific value).

(2) Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category A ICs.

Developer Notes:

For EAL #1 – Enter the radiation monitor(s) that may be used to readily identify when RCS activity levels exceed Technical Specification allowable limits. This EAL may be developed using different methods and sites should use existing capabilities to address it (e.g., development of new capabilities is not required). Examples of existing methods/capabilities include:

- An installed radiation monitor on the Primary Sampling System (AP1000) or Offgas System (ESBWR).
- A hand-held monitor or deployed detector reading with pre-calculated conversion values or readily implementable conversion calculation capability.

The monitor reading values should correspond to an RCS activity level approximately at Technical Specification allowable limits.

For EAL#2 – Developers may reword the EAL to include the reactor coolant activity parameter(s) specified in Technical Specifications and the associated allowable limit(s) (e.g., values for dose equivalent I-131 and gross activity, time-dependent or transient values, etc.). If this approach is selected, all RCS activity allowable limits should be included.

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B

SU4

ECL: Notification of Unusual Event

Initiating Condition: RCS leakage for 15 minutes or longer

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels: (1 or 2 or 3)

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.

(2) RCS (identified leakage [AP1000] or total leakage minus RCS unidentified leakage [ESBWR]) greater than (site-specific value) for 15 minutes or longer.

(3) Leakage from the RCS to a location outside (containment [AP1000] or primary containment [ESBWR]) greater than 25 gpm for 15 minutes or longer.

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

EAL #1 and EAL #2 are focused on a loss of mass from the RCS due to “unidentified leakage", "pressure boundary leakage" or "identified leakage [AP1000] / total leakage [ESBWR]” (as these leakage types are defined in the plant Technical Specifications).

EAL #3 addresses a RCS mass loss not considered either unidentified or (identified [AP1000] or total [ESBWR]) leakage caused by an UNISOLABLE leak through an interfacing system external to the (primary [ESBWR]) containment. These EALs thus apply to leakage into the containment, (a secondary-side system (e.g., steam generator tube leakage) [AP1000]) or a location outside of (primary [ESBWR]) containment.

The leak rate values for each EAL were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). EAL #1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. For AP1000s, an emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated). For ESBWRs, a stuck-open

Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category A or F.

Developer Notes:

EAL #1 – For the site-specific leak rate value, enter the higher of 10 gpm or the value specified in the site Technical Specifications for this type of leakage.

EAL #2 – For the site-specific leak rate value, enter the higher of 25 gpm or the value specified in the site Technical Specifications for this type of leakage.

ECL Assignment Attributes: 3.1.1.A

SU5

ECL: Notification of Unusual Event

Initiating Condition: Automatic or manual (trip [AP1000] / scram [ESBWR]) fails to shut down the reactor

Operating Mode Applicability: Power Operation

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Example Emergency Action Levels: (1 or 2)

(1) a. An automatic (trip [AP1000] / scram [ESBWR]) did not (site-specific indication of reactor shutdown).

AND

b. A subsequent manual action taken at the (site-specific reactor control consoles) is successful in (site-specific indication of reactor shutdown).

(2) a. A manual trip ([AP1000] / scram [ESBWR]) did not (site-specific indication of reactor shutdown).

AND

b. **EITHER** of the following:

1. A subsequent manual action taken at the (site-specific reactor control consoles) is successful in (site-specific indication of reactor shutdown).

OR

2. A subsequent automatic (trip [AP1000R] / scram [ESBWR]) is successful in (site-specific indication of reactor shutdown).

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor (trip [*AP1000*] / scram [*ESBWR*]) that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic (trip [*AP1000R*] / scram [*ESBWR*]) is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor (trip [*AP1000*] / scram [*ESBWR*]), operators will promptly initiate manual actions at the reactor control consoles to shut down the reactor (e.g., initiate a manual reactor (trip [*AP1000*] / scram [*ESBWR*])). If these manual actions are

successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor (trip [*AP1000*] / scram [*ESBWR*]) is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shut down the reactor (e.g., initiate a manual reactor (trip [*AP1000*] / scram [*ESBWR*])) using a different switch). Depending upon several factors, the initial or subsequent effort to manually (trip [*AP1000*] / scram [*ESBWR*]) the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor (trip [*AP1000*] / scram [*ESBWR*]) signal. If a subsequent manual or automatic (trip [*AP1000*] / scram [*ESBWR*]) is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor (trip [*AP1000*] / scram [*ESBWR*])). This action includes the (reactor scram buttons and Reactor Mode Switch [*ESBWR*] / DAS and PDSP reactor trip switches [*AP1000*]). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room (e.g. any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to SHUTDOWN is a manual scram action. [*ESBWR*]

The plant response to the failure of an automatic or manual reactor (trip [*AP1000*] / scram [*ESBWR*]) will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA5. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA5 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor (trip [*AP1000*] / scram [*ESBWR*]) signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied:

- If the signal causes a plant transient that should have included an automatic reactor (trip [*AP1000*] / scram [*ESBWR*]) and the RPS fails to automatically shut down the reactor, this IC and the EALs are applicable, and must be evaluated.

- If the signal does not cause a plant transient and the (trip [*AP1000*] / scram [*ESBWR*]) failure is determined through other means (e.g., assessment of test results), this IC and the EALs are not applicable and no classification is warranted.

Developer Notes:

This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. An AP1000 with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, the IC is also applicable in Startup Mode.

Developers should include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level). For AP1000s that implement the WOG EPGs this value is the Red Path CSFST entry condition, typically 5% rated thermal power. For ESBWRs, this value is the APRM downscale setpoint, also typically 5% rated thermal power. In both cases these power levels represent approximately decay heat levels following a reactor (trip [*AP1000*] / scram [*ESBWR*]).

The term “reactor control consoles” may be replaced with the appropriate site-specific term(s) (e.g., Main Control Console [*ESBWR*] or Control Room Workstations [*AP1000*]).

ECL Assignment Attributes: 3.1.1.A

SU6

ECL: Notification of Unusual Event

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

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels: (1 or 2 or 3)

(1) Loss of **ALL** of the following onsite communication methods:

(site-specific list of communications methods)

(2) Loss of **ALL** of the following ORO communications methods:

(site-specific list of communications methods)

(3) Loss of **ALL** of the following NRC communications methods:

(site-specific list of communications methods)

Basis:

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

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

EAL #1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL #2 addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are (see Developer Notes).

EAL #3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

Developer Notes:

EAL #1 - The "site-specific list of communications methods" should include all communications methods used for routine plant communications (e.g., commercial or site telephones, page-party systems, radios, etc.). This listing should include installed plant equipment and components, and not items owned and maintained by individuals.

EAL #2 - The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to OROs as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. Example methods are ring-down/dedicated telephone lines, commercial telephone lines, radios, satellite telephones and internet-based communications technology.

In the Basis section, insert the site-specific listing of the OROs requiring notification of an emergency declaration from the Control Room in accordance with the site Emergency Plan, and typically within 15 minutes.

EAL #3 – The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to the NRC as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. These methods are typically the dedicated Emergency Notification System (ENS) telephone line and commercial telephone lines.

ECL Assignment Attributes: 3.1.1.C

SU7

ECL: Notification of Unusual Event

Initiating Condition: Failure to isolate containment or loss of containment pressure control. [*AP1000*]

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Safe Shutdown

Example Emergency Action Levels: (1 or 2)

(1) a. Failure of containment to isolate when required by an actuation signal.

AND

b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal.

(2) a. Containment pressure greater than (site-specific pressure setpoint).

AND

b. PCS does not actuate and perform its safety function per design for 15 minutes or longer.

Basis:

This IC addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For EAL #1, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., low pressurizer level or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

EAL #2 addresses a condition where containment pressure is greater than the setpoint at which the Passive Containment Cooling System (PCS) is designed to automatically actuate per design. The 15-minute criterion is included to allow operators time to manually align PCS components that may not have automatically actuated, if possible. The inability to actuate the required equipment and perform its safety function indicates that containment heat removal/depressurization systems are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

Developer Notes:

Enter the site-specific pressure setpoint value that actuates PCS. If desired, specific condition indications such as parameter values can also be entered that indicate a failure of PCS to actuate per design.

ECL Assignment Attributes: 3.1.1.A

SA1

ECL: Alert

Initiating Condition: Loss of all ability to charge at least one (site-specific Class 1E DC battery) for 60 minutes or longer

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels:

Note: The Emergency Director should declare the Alert promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.

(1) Loss of all ability to charge at least one (site-specific Class 1E 24-hr DC battery) for 60 minutes or longer

Basis:

This IC and the associated threshold is intended to provide an escalation from IC SU1. Prolonged de-energization of the busses that provide power to the Class 1E 24-hr battery chargers reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of DC power even though no operator action is required for 72 hours.

60 minutes was selected to allow power restoration procedures to be effective.

The Class 1E (safety-related) 24-hr DC system provides electrical power for safety-related and vital control and monitoring instrumentation loads. The offsite AC power system normally supplies power for the unit in cold shutdown, refueling, and defueled conditions. Both the normal offsite and standby onsite AC power systems are non-Class 1E with no Technical Specification requirements. All safety-related functions associated with the unit in cold shutdown and refueling are provided by the safety-related onsite Class 1E DC power systems including the 120V Vital AC power system supplied from the batteries powering the safety-related DC buses through inverters. The Passive ALWRs also have standby diesel generators that are not safety-related. Storage batteries are the safety-related power source for Class 1E electric power. However, this non-Class 1E AC power is required to charge the safety-related DC batteries and maintain long-term safety-related DC bus voltage.

Escalation of the emergency classification level would be via IC SS1.

Developer Notes:

Buses ECS-ES-1 and 2 for AP1000 or the PIP A3 and B3 buses for ESBWR are the source of non-Class 1E AC power to the Class 1E DC battery chargers.

ECL Assignment Attributes: 3.1.2.B

SA2

ECL: Alert

Initiating Condition: Partial loss of monitoring or control functions for 15 minutes or longer.

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels:

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) UNPLANNED loss of the ability to monitor or control one or more of the following key safety functions from the Main Control Room for 15 minutes or longer:

- Reactivity control
- Core cooling [*AP1000*] / RPV water level [*ESBWR*]
- RCS heat removal
- Spent Fuel Pool Level/Temperature Control

Basis:

This IC recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the control and indication systems. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

This IC recognizes the challenge to the Control Room staff to monitor and control the plant due to partial loss of normal and safety indication and monitoring systems. Escalation to a Site Area Emergency will be via SS2 if a complete loss of control and indication occurs. Declaration of the Alert will provide the Control Room staff with additional personnel to assist in monitoring alternative indications, manipulating equipment and restoring the systems to full capability. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems due to an inadvertent loss.

<u>AP1000</u>

Systems that provide monitoring and control capability include:

The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shutdown the plant, and to maintain the plant in a safe shutdown condition.

The Plant Control System (PLS) is a nonsafety-related system and includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy

into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine.

The Data Display and Processing System (DDS) (Plant Computer) is comprised of a set of graphics workstations that obtains its inputs from real-time data networks and delivers its output to the network for plant operators and other users.

The Diverse Actuation System (DAS) is a nonsafety-related system available to ensure monitoring and control capability as well as providing an alternative means of initiating a reactor trip and selected engineering safety features.

ESBWR

The Essential Distributed Control and Information System (E-DCIS) provides redundant data communications networks to support the monitoring and control of interfacing safety-related control and instrumentation systems. The system includes electrical devices and circuitry that connect field sensors, display devices, controllers, power supplies, and actuators, which are part of these safety-related systems. The E-DCIS also includes any associated data acquisition and communications software, if required, to support its distribution function of data and control. The system processes data from safety-related systems and safety-related trip or initiation data strictly through E-DCIS, while nonsafety-related data is processed through the Non-Essential DCIS

The E-DCIS provides the data processing and transmission network that encompasses the four independent and separate data multiplexing divisions 1, 2, 3, and 4, corresponding to the four divisions of safety-related electrical and I&C equipment. Total loss of E-DCIS would result in escalation to SS7

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

Escalation of the emergency classification level would be via ICs FS1 or IC AS1.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

SA5

ECL: Alert

Initiating Condition: Automatic or manual (trip [AP1000] / scram [ESBWR]) fails to shut down the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

Operating Mode Applicability: Power Operation

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Example Emergency Action Levels:

(1) a. An automatic or manual (trip [AP1000] / scram [ESBWR]) did not reduce reactor power below (site-specific indication of reactor shutdown).

AND

b. Manual actions taken at the (site-specific reactor control consoles) are not successful in reducing reactor power below (site-specific indication of reactor shutdown).

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor (trip [*AP1000*] / scram [*ESBWR*]) that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shut down the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shut down by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor (trip [*AP1000*] / scram [*ESBWR*])). This action includes the (reactor scram buttons and Reactor Mode Switch [*ESBWR*] / DAS and PDSP reactor trip switches [*AP1000*]). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (e.g., locally opening breakers). Actions taken at back-panels or other locations within the Control Room (e.g. any location outside the Control Room, are not considered to be "at the reactor control consoles").

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [*ESBWR*]

The plant response to the failure of an automatic or manual reactor (trip [*AP1000*] / scram [*ESBWR*]) will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling [AP1000] / RPV water level [ESBWR] or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS5. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS5 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Developer Notes:

This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. An AP1000 with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode.

Developers should include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level). For AP1000s,this value is the Red Path CSFST entry condition, typically 5% rated thermal power. For ESBWRs, this value is the APRM downscale setpoint, also typcally 5% rated thermal power. In both cases these power levels represent approximately decay heat levels following a reactor (trip [*AP1000*] / scram [*ESBWR*]).

The term “reactor control consoles” may be replaced with the appropriate site-specific term(s) (e.g., Main Control Console [*ESBWR*] or Control Room Work Stations [*AP1000*].).

ECL Assignment Attributes: 3.1.2.B

SA8

ECL: Alert

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

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels:

(1) a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- (site-specific hazards)
- Other events with similar hazard characteristics as determined by the Shift Manager

AND

b **EITHER** of the following:

1. Event damage has caused indications of degraded performance in at least one division of a SAFETY SYSTEM needed for the current operating mode.

   OR

2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.

Basis:

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

EAL 1.b.1 addresses damage to a SAFETY SYSTEM division that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM division.

EAL 1.b.2 addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or AS1.

Developer Notes:

For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche or tsunami).

Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant divisions of equipment in accordance with site-specific design criteria.

ECL Assignment Attributes: 3.1.2.B

SS1

ECL: Site Area Emergency

Initiating Condition: Loss of all required (site-specific Class 1E DC power) or (site-specific Class 1E UPS bus power) for 15 minutes or longer

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Indicated voltage is less than (site-specific bus voltage value) on ALL required (site-specific Class 1E 24-hr. DC buses) for 15 minutes or longer.

(2) Loss of power to ALL (site-specific Class 1E UPS buses) for 15 minutes or longer.

Basis:

Loss of all 24-hr. 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.

This IC addresses a total loss of Class 1E 24-hr. DC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Threshold 1 (site-specific minimum bus voltage) is the minimum bus voltage necessary for the operation of safety-related instrumentation and controls. This voltage value incorporates a margin significantly longer than the allowed 15 minutes of operation before the onset of inability to operate those loads.

Threshold 2 addresses an event that results in de-energizing all UPS busses. This condition would result in degraded capability to monitor and control the unit from the Main Control Room resulting in the need for additional personnel to manage the event.

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

Escalation of the emergency classification level would be via ICs AG1 or FG1.

Developer Notes:

The site-specific Class 1E DC bus voltage should be based on the minimum battery terminal voltage at the end of the discharge period.

The site-specific Class 1E UPS buses are the 120 VAC safety-related buses

ECL Assignment Attributes: 3.1.3.B

SS2

ECL: Site Area Emergency

Initiating Condition: Complete loss of monitoring or control functions for 15 minutes or longer

Operating Mode Applicability: Power Operation, Startup, Hot Shutdown/Hot Standby, Safe Shutdown

Example Emergency Action Levels:

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Loss of ability to monitor or control ALL of the following key safety functions from the Main Control Room for 15 minutes or longer:

- Reactivity control
- Core cooling [*AP1000*] / RPV water level [*ESBWR*]
- RCS heat removal
- Spent Fuel Pool Level/Temperature Control

Basis:

This IC recognizes the inability of the Control Room staff to monitor and control the plant due to loss of normal and safety indication and monitoring systems, and diverse indication and control systems that allow the operators to monitor and safely shutdown the plant. A Site Area Emergency is considered to exist if the Control Room staff cannot monitor and control safety functions needed for protection of the public.

This IC recognizes the challenge to the Control Room staff to monitor and control the plant due to a complete loss of normal and safety indication and monitoring systems. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems due to an inadvertent loss.

<u>AP1000</u>

Systems that provide monitoring and control capability include:

The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shutdown the plant, and to maintain the plant in a safe shutdown condition.

The Plant Control System (PLS) is a nonsafety-related system and includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine.

The Data Display and Processing System (DDS) (Plant Computer) is comprised of a set of graphics workstations that obtains its inputs from real-time data networks and delivers its output to the network for plant operators and other users.

The Diverse Actuation System (DAS) is a nonsafety-related system available to ensure monitoring and control capability as well as providing an alternative means of initiating a reactor trip and selected engineering safety features.

ESBWR

The Essential Distributed Control and Information System (E-DCIS) provides redundant data communications networks to support the monitoring and control of interfacing safety-related control and instrumentation systems. The system includes electrical devices and circuitry that connect field sensors, display devices, controllers, power supplies, and actuators, which are part of these safety-related systems. The E-DCIS also includes any associated data acquisition and communications software, if required, to support its distribution function of data and control. The system processes data from safety-related systems and safety-related trip or initiation data strictly through E-DCIS, while nonsafety-related data is processed through the Non-Essential DCIS

The E-DCIS provides the data processing and transmission network that encompasses the four independent and separate data multiplexing divisions 1, 2, 3, and 4, corresponding to the four divisions of safety-related electrical and I&C equipment.

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

Escalation of the emergency classification level would be via ICs FS1 or IC AS1.

Developer Notes:

ECL Assignment Attributes: 3.1.3.B

SS5

ECL: Site Area Emergency

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

Operating Mode Applicability: Power Operation

Example Emergency Action Levels:

(1) a. An automatic or manual (trip [AP1000] / scram [ESBWR]) did not (site-specific indication of reactor shutdown).

AND

b. ALL manual actions to shutdown the reactor have been unsuccessful.

AND

c. **EITHER** of the following conditions exist:

- (Site-specific indication of an inability to adequately remove heat from the core)
- (Site-specific indication of an inability to adequately remove heat from the RCS)

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor (trip [*AP1000*] / scram [*ESBWR*]) that results in a reactor shutdown, all subsequent operator actions to manually shut down the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via IC AG1 or FG1.

Developer Notes:

This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. An AP1000 with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode.

Developers should include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level). For AP1000s, this value is the Red Path CSFST entry condition, typically 5% rated thermal power. For ESBWRs, this value is the APRM downscale setpoint, also typcally 5% rated thermal power. In both cases these power levels represent approximately decay heat levels following a reactor (trip [*AP1000*] / scram [*ESBWR*]).

Site-specific indication of an inability to adequately remove heat from the core:

[*ESBWR*] – Reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (as described in the EOP bases).

[*AP1000*] – Insert core exit thermocouple temperatures greater than 1,200^{o}F.

Site-specific indication of an inability to adequately remove heat from the RCS:

[*ESBWR*] - Use the Heat Capacity Temperature Limit. This addresses the inability to reject RCS heat to the main condenser and, as a result, heat addition to the suppression pool has raised pool temperatures to unacceptable levels for the existing suppression pool water level and RPV pressure.

[*AP1000*] - Insert site-specific parameters associated with inadequate RCS heat removal via the steam generators and PRHR. These parameters should be identical to those used for the Inadequate Heat Removal threshold Fuel Clad Barrier Potential Loss 2.B and threshold RCS Barrier Potential Loss 2.A in the AP1000 EAL Fission Product Barrier Table.

ECL Assignment Attributes: 3.1.3.B

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC .... Alternating Current
ADS .... Automatic Depressurization System
ALWR .... Advanced Light Water Reactor
AOP .... Abnormal Operating Procedure
APRM .... Average Power Range Meter
ARI .... Automatic Rod Insertion
ARMS .... Area Radiation Monitoring System
ATWS .... Anticipated Transient Without Scram
B&W .... Babcock and Wilcox
BIIT .... Boron Injection Initiation Temperature
BWR .... Boiling Water Reactor
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CMT .... Core Makeup Tanks
CTMT/CNMT .... Containment
CSF .... Critical Safety Function
CSFST .... Critical Safety Function Status Tree
CVS .... Chemical Volume Control System
DAS .... Diverse Actuation System
DBA .... Design Basis Accident
DC .... Direct Current
DCD .... Design Control Document
DDS .... Data Display and Processing System
DG .... Non-Class 1-E Diesel Generator System
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
ECS .... Main AC Power System
EDS .... Non 1-E DC and UPS
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
EPIP .... Emergency Plan Implementing Procedure
EPRI .... Electric Power Research Institute
ERG .... Emergency Response Guideline
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GDCS .... Gravity Driven Cooling System
GE .... General Emergency
HCTL .... Heat Capacity Temperature Limit
HSI .... Human System Interface
HX .... Heat Exchanger
IC .... Initiating Condition
ICS .... Isolation Condenser System
ID .... Inside Diameter

IDS	Class 1-E DC and UPS Power System
IRWST	In-containment Refueling Water Storage Tank
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI	Independent Spent Fuel Storage Installation
Keff	Effective Neutron Multiplication Factor
LCO	Limiting Condition of Operation
LD&IS	Leak Detection & Isolation System
LOCA	Loss of Coolant Accident
MCR	Main Control Room
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSSV	Main Steam Safety Valve
mR, mRem, mrem, mREM	milli-Roentgen Equivalent Man
MW	Megawatt
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NORAD	North American Aerospace Defense Command
(NO)UE	(Notification Of) Unusual Event
NUMARC[8]	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
OCA	Owner Controlled Area
ODCM/ODAM	Offsite Dose Calculation (Assessment) Manual
ORO	Off-site Response Organization
PA	Protected Area
PACS	Priority Actuation and Control System
PAG	Protective Action Guideline
PCS	Passive Containment Cooling System
PDSP	Primary Dedicated Safety System
PICS	Process Information and Control System
PORV	Power Operated Relief Valve
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PRHR	Passive Residual Heat Removal
PWR	Pressurized Water Reactor
PS	Protection System
PSIG	Pounds per Square Inch Gauge
PXS	Passive Core Cooling System
QDPS	Qualified Data Processing System
R	Roentgen
RCC	Reactor Control Console
RCS	Reactor Coolant System
Rem, rem, REM	Roentgen Equivalent Man
RETS	Radiological Effluent Technical Specifications
RHR	Residual heat Removal
RNS	Normal Residual Heat Removal

[8] NUMARC was a predecessor organization of the Nuclear Energy Institute (NEI).

RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RVLIS .... Reactor Vessel Level Instrumentation System
RWCU .... Reactor Water Cleanup
RSW .... Remote Shutdown workstation
SAG .... Severe Accident Guideline
SAR .... Safety Analysis Report
SAS .... Safety Automation System
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SDP .... Secondary Dedicated Panel
SFP .... Spent Fuel Pool
SG .... Steam Generator
SI .... Safety Injection
SICS .... Safety Information and Control System
SPDS .... Safety Parameter Display System
SRM .... Source Range Monitor
SRO .... Senior Reactor Operator
SRV .... Safety Relief Valve
TEDE .... Total Effective Dose Equivalent
TAF .... Top of Active Fuel
TSC .... Technical Support Center
UFSAR .... Updated Final Safety Analysis Report
VBS .... Nuclear Island Non-radioactive Ventilation System
VES .... Main Control Room Emergency Ventilation
WOG .... Westinghouse Owners Group
WPIS .... Wall Panel Indication System

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APPENDIX B – DEFINITIONS

The following definitions are taken from Title 10, Code of Federal Regulations, and related regulatory guidance documents.

Alert: Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.

General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Notification of Unusual Event (NOUE)[9]: 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.

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

The following are key terms necessary for overall understanding the NEI 99-01 emergency classification scheme.

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

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

- Notification of Unusual Event (NOUE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

[9] This term is sometimes shortened to Unusual Event (UE) or other similar site-specific terminology.

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

Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

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

CONFINEMENT BOUNDARY: (Insert a site-specific definition for this term.) **Developer Note** – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

CONTAINMENT CLOSURE: (Insert a site-specific definition for this term.) **Developer Note** – The plant Technical Specification and site-specific procedurally defined conditions or actions taken to secure containment (primary or secondary for ESBWR) and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

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

FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized. **Developer Note** – This term is applicable to AP1000s only.

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

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

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

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

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

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.

OWNER CONTROLLED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the site property owned by, or otherwise under the control of, the licensee. In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan.

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

PROTECTED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY: (Insert a site-specific definition for this term.) **Developer Note** – This description should include all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.

RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection. **Developer Note** – This term is applicable to PWRs only.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related. **Developer Note** – This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.

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.

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

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

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