NEI 99-01 [Revision 6]

Methodology for Development of Emergency Action Levels

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Nuclear Energy Institute

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January 2011

Nuclear Energy Institute, 1776 I Street N. W., Suite 400, Washington D.C. (202.739.8000)

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This document was prepared by the Nuclear Energy Institute (NEI) Emergency Action Level (EAL) Task Force.

NEI Chairperson: Martin Hug

Industry Lead: Keith Kemper, Exelon Nuclear/Corporate

Preparation Team

Team Leader**:** David Young – NextEra Energy/Seabrook Station
Larry Baker – Exelon Nuclear/Corporate
Craig Banner – PSEG Nuclear/Salem and Hope Creek Nuclear Generating Stations
John Egdorf – Dominion Generation/Kewaunee Power Station
Jack Lewis – Entergy Nuclear/Corporate
C. Kelly Walker – Operations Support Services, Inc.

Review Team

Chris Boone – Southern Nuclear/Corporate
Kent Crocker – Progress Energy/Brunswick Nuclear Plant
Don Crowl – Duke Energy/Corporate
Gerry Holthaus – Xcel Energy/Monticello Nuclear Generating Plant
John Kaminski – Constellation Energy/Nine Mile Point Nuclear Station
Walt Lee – TVA Nuclear/Corporate
Jay Maisler – Enercon Services, Inc.
Scott McCain - Emergency Planning Technical Consultants, Inc.
Ken Meade – FENOC/Corporate
David Stobaugh – EP Consulting, LLC
Nick Turner – STARS/Callaway Plant
Maureen Zawalick – STARS/Diablo Canyon Power Plant

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) 99-01 is to provide guidance to nuclear power plant operators 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.

NEI 99-01 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. The ICs, EALs and thresholds presented in this document are not intended to be used "as-is"; rather, scheme developers will need to use this generic information and related development guidance to create their site-specific scheme.

Properly implemented, the guidance in NEI 99-01 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 99-01 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. In addition, NEI maintains a standing task force to address issues and enhancements related to the NEI 99-01 methodology. Task force members are a valuable resource that developers may consult with questions concerning implementation of this document.

Finally, State and local requirements associated with an emergency classification scheme are not reflected in this generic guidance. Incorporation of these requirements, if any, should be performed on a case-by-case basis in conjunction with appropriate ORO personnel.

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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. The following excerpts from Title 10 describe the requirements applicable to the purpose, content and use of an emergency classification scheme, and other related regulatory-driven processes. The intent here is to make the reader aware of these particular regulations and to provide context for the key terminology provided in Section 3.0 of this document.

1.1.1 10 CFR § 50.2 – "Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary

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

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable."

1.1.2 10 CFR § 50.47 (a)(1)(i) – " . . . no initial operating license for a nuclear power reactor will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency."

1.1.3 10 CFR § 50.47(b) – "The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:"

1.1.3.1 10 CFR § 50.47(b)(4) – "A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures."

1.1.4 10 CFR § 50, Appendix E – "This appendix establishes minimum requirements for emergency plans for use in attaining an acceptable state of emergency preparedness."

1.1.5 10 CFR § 50, Appendix E, IV, *Content of Emergency Plans* – "The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, i.e., organization for coping with radiation emergencies, assessment action, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, and recovery. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license shall contain information needed to demonstrate compliance with the standards described in § 50.47(b), and they will be evaluated against those standards."

1.1.5.1 10 CFR § 50, Appendix E, IV.B, *Assessment Actions* – "The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in- plant conditions and instrumentation in addition to onsite and offsite monitoring. These action levels must include hostile action events expected to adversely affect the plant."

1.1.5.2 10 CFR § 50, Appendix E, IV.C, *Activation of Emergency Organization*

(1) "The entire spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as the pressure in containment and the response of the Emergency Core Cooling System) for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) notification of unusual events, (2) alert, (3) site area emergency, and (4) general emergency. These classes are further discussed in NUREG-0654; FEMA-REP- 1.

(2) An applicant or a licensee shall 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. This 15-minute criterion must not be construed as a grace period to attempt to restore plant conditions to avoid declaring an emergency action due to an EAL that has been exceeded. This 15-minute criterion must not be construed as preventing implementation of response actions deemed by the licensee to be necessary to protect health and safety provided that any delay in classification does not deny the State and local authorities the opportunity to implement measures necessary to protect

the public health and safety."

1.1.6 10 CFR § 50.72(a), *General Requirements* – "(1) Each nuclear power reactor licensee . . . shall notify the NRC Operations Center via the Emergency Notification System of:

(i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan; or
(ii) Those non-emergency events specified in paragraph (b) of this section that occurred within three years of the date of discovery."

Changes to an emergency classification scheme are governed by the requirements of 10 CFR § 50, Appendix E, Section B, and 10 CFR § 50.54(q), and related NRC requirements and guidance (e.g., Regulatory Guides, Regulatory Issue Summaries, etc.). A full discussion of scheme change requirements is beyond the scope of this document.

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. [Refer to Section 3.1.1, *Immediate Notifications*]
- Regulatory Guide 1.101, *Emergency Response Planning and Preparedness for Nuclear Power Reactors*

1.2 PERMANENTLY DEFUELED STATION

NEI 99-01 provides guidance for an emergency classification scheme applicable to a permanently defueled station. This is a station that generated spent fuel under a 10 CFR § 50 license, has permanently ceased operations and will store the spent fuel onsite for an extended period of time. The emergency classification levels applicable to this type of station are consistent with the requirements of 10 CFR § 50 and the guidance in NUREG 0654/FEMA-REP-1.

In order to relax the emergency plan requirements applicable to an operating station, the owner of a permanently defueled station must demonstrate that no credible event can result in a significant radiological release beyond the site boundary. It is expected that this verification will confirm that the source term and motive force available in the permanently defueled condition are insufficient to warrant classifications of a Site Area Emergency or General Emergency. Therefore, the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) applicable to a permanently defueled station may result in either a Notification of Unusual Event (NOUE) or an Alert Classification.

The generic ICs and EALs are presented in Section 8, *Permanently Defueled Station ICs/EALs*. Refer to Appendix B of this document for additional information.

1.3 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

NEI 99-01 provides guidance for an emergency classification scheme applicable to an ISFSI. It may be used by licensees who elect to meet the requirements of 10 CFR § 72.32 via a site emergency plan developed and approved under 10 CFR § 50. 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; therefore, 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 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 Effective Dose Equivalent.

The expectations for 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 (i.e., to provide assistance if requested). Even with regard to activation of a licensee's Emergency Response Organization (ERO), the ERO for a 10 CFR § 72.32 emergency plan is not that prescribed under a 10 CFR § 50.47 emergency plan (e.g., no emergency technical support). Consequently, the "Alerts" contemplated by 10 CFR § 72.32, have been classified as NOUEs in the NEI 99-01 methodology. To do otherwise could lead to an inappropriate response posture on the part of offsite response organizations.

It is expected that a NOUE will be the highest emergency classification level necessary to respond to any credible event affecting an ISFSI.

The generic ICs and EALs are presented in Section 9, ISFSI ICs/EALs. These ICs and EALs are not applicable to stand-alone ISFSIs, Monitored Retrievable Storage Facilities (MRSF), or ISFSIs that may process and/or repackage spent fuel.

1.4 REGULATORY EXPECTATIONS CONCERNING EMERGENCY CLASSIFICATION SCHEME CHANGES

Regulatory Issue Summary (RIS) 2003-18, and its associated supplements, discuss NRC expectations concerning implementation of NEI 99-01. Specifically, the RIS provides clarification of NRC staff expectations for implementing an emergency classification scheme change. A scheme change occurs when a site revises its emergency classification scheme to rely on (i.e., be based upon) a new basis document and/or revision of that basis document.

RIS 2003-18 provides NRC expectations concerning alignment of a site-specific scheme with the generic guidance presented in NEI 99-01 and discusses associated regulatory review requirements. Developers of an emergency classification scheme should become familiar with the guidance in RIS 2003-18. Questions concerning this guidance may be directed to the NEI EP staff.

2 KEY TERMINOLOGY RELATED TO NEI 99-01 GUIDANCE

The following key terminology is used in the NEI 99-01 methodology.

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 seriousness, are called:

- 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.

[1] This term is sometimes shortened to Unusual Event (UE) or other similar site-specific terminology.

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 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 meets the definition of one of the four emergency classification levels based on potential or actual effects or consequences.

Discussion: In NUREG-0654, the NRC introduced, but does not define, the term "initiating condition." Since the term is commonly used in nuclear power plant emergency planning, the definition above has been developed and combines both regulatory intent and the greatest degree of common usage among nuclear power plants.

An IC describes a unique 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 explosion within the Protected Area) or the status of one or more fission product barriers (e.g., loss of the RCS barrier).

Considerations for assigning 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 a plant IC that places the plant in a given emergency classification level. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency classification level.

Discussion: The term "emergency action level" has been defined by example in the regulations, as noted in the above discussion concerning regulatory background. There are times when an EAL will be a threshold point on a measurable continuous function, such as a reactor coolant system leak that has exceeded a Technical Specification limit. At other times, the EAL and the IC will coincide, both identified by a discrete event or condition that places the plant in a particular emergency classification level.

2.4 FISSION PRODUCT BARRIER THRESHOLD:

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of one or more of the fission product barriers. 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 emergency classification (Refer to Section 10 Fission Product Barrier ICs/EALs).

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

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

2.5 SAFETY-RELATED:

A system, structure or component relied upon to remain functional during and following a design basis event in order to protect the integrity of the reactor coolant pressure boundary; shut down the reactor and maintain it in a safe shutdown condition; or prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11.

Discussion: Typically, a site’s Final Safety Analysis Report, as updated, will define the plant-specific systems, structures, and components that are safety-related. In addition, § 50.36 contains the following criteria:

“A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

These criteria, in conjunction with a review of plant-specific Technical Specifications, may also be used to identify safety-related systems, structures, and components. Because this term is used in numerous ICs and bases discussions, it is also included in Appendix D, Definitions.

3 DESIGN OF THE NEI 99-01 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 99-01 emergency classification scheme attempts to strike an appropriate balance between reasonably anticipated event or condition consequences, potential accident trajectories, and risk avoidance or minimization.

As noted in Section 1, numerous types of "non-emergency events" are required to be reported to the US Nuclear Regulatory Commission (NRC) staff in accordance with 10 CFR § 50.72. Clarification of these requirements and example events are provided in NUREG-1022. The NEI 99-01 emergency classification scheme is designed to not overlap classifiable emergency events with non-emergency reportable events, i.e., no non-emergency reportable event would require an emergency declaration.

In order to align Initiating Conditions (ICs) with the appropriate ECL, it is necessary to determine the attributes of each ECL. The goal of this process is to answer the question, "What types of events or conditions should be placed under each ECL?" The following sources provided information and context for the development 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
- Radiological Effluent Technical Specifications (RETS)/Offsite Dose Calculation Manual (ODCM) radiological release limits
- Review of selected Updated Final Safety Analysis Report (UFSAR) accident analyses
- Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs)
- NUREG 0654, Appendix 1, *Emergency Action Level Guidelines for Nuclear Power Plants*
- Industry Operating Experience
- Input from industry subject matter experts and NRC staff members

The attributes of each ECL are presented below.

3.1.1 Notification of Unusual Event (NOUE)

A Notification of Unusual Event represents an event or condition that involves:

(A) A precursor to a more significant event or condition (e.g., inability to meet certain requirements in operating procedures, an event or SECURITY CONDITION that poses a threat to plant personnel or SAFETY-RELATED equipment, etc.).

(B) The plant being outside the safety envelope defined by Technical Specifications (Limiting Conditions for Operation and associated Action Statements).

(C) The exposure of SAFETY-RELATED systems, structures or components to conditions beyond design limits.

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

3.1.2 Alert

An Alert represents 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) A precursor event or condition that may lead to a loss or potential loss of the fuel clad or RCS fission product barrier. Precursor events and conditions of this type include those resulting in VISIBLE DAMAGE to a SAFETY-RELATED structure or area; damage sufficient to degrade the performance of multiple SAFETY-RELATED trains or systems; or a Control Room evacuation.

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

(D) A HOSTILE ACTION occurring within the OWNER CONTROLLED AREA or directed at an Independent Spent Fuel Storage Installation (ISFSI) located outside the plant PROTECTED AREA.

3.1.3 Site Area Emergency

A Site Area Emergency represents an event or condition that involves:

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

(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-RELATED systems.

(C) A release of radioactive materials to the environment associated with the loss of two fission product barriers; offsite doses will not exceed any 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 represents 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 subsequent loss of containment integrity.

(C) A release of radioactive materials to the environment associated with the loss of all three fission product barriers; offsite doses will exceed 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

The assignment of ECLs also considered insights from several plant-specific probabilistic safety assessments (PSA - also known as probabilistic risk assessment, PRA). PSAs were reviewed to determine the risk associated with particular emergency conditions. Some generic insights from this review included:

1. Accident sequences involving a prolonged loss of all AC power are significant contributors to core damage frequency at many Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). For this reason, a loss of all AC power for greater than 15 minutes, with the plant at or above Hot Shutdown, was assigned an ECL of Site Area Emergency. Precursor events to a loss of all AC power were also included as an Unusual Event and an Alert.

   A station blackout coping analyses performed in response to 10 CFR § 50.63 and Regulatory Guide 1.155, *Station Blackout*, may be used to determine a time-based criterion to demarcate between a Site Area Emergency and a General Emergency. The time dimension is critical to a properly anticipatory emergency declaration since the goal is to maximize the time available for State and local officials to develop and implement offsite protective actions.

2. For severe core damage events, uncertainties exist in phenomena important to accident progressions leading to containment failure. Because of these uncertainties, predicting the status of containment integrity may be difficult under severe accident conditions. This is why maintaining containment integrity alone following sequences leading to severe core damage is an insufficient basis for not escalating to a General Emergency.

3. PSAs indicated that leading contributors to latent fatalities were sequences involving a containment bypass, a large Loss of Coolant Accident (LOCA) with early containment failure, a Station Blackout longer than 6 hours (e.g., blackout-induced LOCA), and a reactor coolant pump seal failure. The generic EAL methodology needs to be sufficiently rigorous to address these sequences in a timely fashion.

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

The NEI 99-01 methodology makes use of symptom-based, event-based, and barrier-based ICs and EALs. The background of each type is discussed below.

Symptom-based ICs and EALs are parameters or conditions that are measurable over some continuous spectrum using plant instrumentation (e.g., core temperature, reactor coolant level, containment pressure, etc.). When one or more of these parameters or conditions are off-normal, reactor operators will implement procedures to identify the probable cause(s) and take corrective action. The level of seriousness that these symptoms indicate depends on the degree to which they have exceeded technical specification or plant design limits, the occurrence of other contemporaneous events or conditions, and the degree to which operators can regain control of the plant function and bring it back to safe and expected levels.

Event-based ICs/EALs define occurrences with potential or actual safety significance, such as the failure of a high pressure safety injection pump, a safety valve failure, or a loss of SAFETY-RELATED electric power to some part of the plant. The range of seriousness of these events is dependent on the location, number of contemporaneous events, remaining plant safety margin, etc.

Barrier-based ICs and EALs refer 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. Barrier-based EALs are a subset of symptom-based EALs that deal exclusively with symptoms indicating fission product barrier challenges.

Some barrier-based EAL thresholds include indications arising from the implementation of Emergency Operating Procedures (EOPs).

Observable indications for a NOUE or an Alert can be events (e.g., natural phenomena), symptoms (e.g., high temperature, low water level), or barrier-related (e.g., challenge to fission product barrier). As the ECL escalates to a Site Area Emergency and General Emergency, the initiating event(s) leading to the emergency classification becomes less important relative to the resulting symptoms (including those associated with challenges to fission product barriers). Thus, EALs for these emergency classification levels are primarily symptom and barrier-based.

General Emergency conditions would be accompanied by increased uncertainties in system or structure (e.g. containment) response and accident progression. To better assure timely classification and notification, EALs in this category are primarily expressed in terms of plant safety function status and parameters, with a secondary reliance on dose projections and field monitoring.

A large source-term within the containment may result in an EPA PAG being exceeded offsite due to expected and allowable containment leakage. The risk of exceeding a PAG increases with any challenges to the containment fission product barrier. 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.

NEI 99-01 emergency classification scheme was developed recognizing that the applicability and mix of ICs and EALs will vary with plant mode. For example, some symptom-based ICs and EALs are available for assessment only in normal startup, operating or hot shutdown modes of operation when all fission product barriers are in place, and plant instrumentation and safety systems are fully operational as required by Technical Specifications. In 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.

3.3 NSSS DESIGN DIFFERENCES

The NEI 99-01 emergency classification scheme accounts for the design differences between PWRs and BWRs by specifying EALs unique to each type of Nuclear Steam Supply System (NSSS). There are also significant design differences among PWR NSSSs. Guidance is provided to aid in the development of EALs appropriate to different PWR NSSS types.

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 to the intent of generic EALs as possible within the constraints imposed by the plant design and operating characteristics.

The guidance in NEI 99-01 is not applicable to advanced passive light water reactor designs. An Emergency Classification Scheme for this type of plant should be developed in accordance with NEI 07-01, Methodology for Development of Emergency Action Levels, Advanced Passive Light Water Reactors.

3.4 WESTINGHOUSE CRITICAL SAFETY FUNCTIONS

The Emergency Response Guidelines (ERGs) developed by the Westinghouse Owners Group (WOG) define a set of Critical Safety Functions that guides the development and implementation of EOPs. The EOPs are designed 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). The guidance in NEI 99-01 allows for the optional 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.

3.5 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
- C - Cold Shutdown / Refueling System Malfunction
- D - Permanently Defueled Station
- E - Independent Spent Fuel Storage Installation (ISFSI)
- F - Fission Product Barrier
- H - Hazards and Other Conditions Affecting Plant Safety
- S - System Malfunction

Each Recognition Category section contains a matrix listing the ICs and their associated emergency classification levels. These matrices provide the reader with an overview of how the ICs are logically related under each emergency classification level.

The following information and guidance is provided for each IC:

- **Initiating Condition –** Specifies the assigned ECL and states the generic description of the emergency event or condition. It is possible that a generic IC cannot be used because the intent cannot be met (e.g., the IC is incompatible with the plant location or design). The developer will need to clearly document the basis for not incorporating the IC into the site-specific scheme.

- **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). Note that Permanently Defueled Station and ISFSI IC/EALs have no mode applicability.

- **Example Emergency Action Level(s) –** Provides examples of reports and indications that are considered to meet the intent of the IC. For Recognition Category F, the fission product barrier-based EALs and thresholds are presented in tables applicable to BWRs and PWRs, 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 EALs and thresholds, and supports more accurate dynamic assessments.

  Developers should address each example EAL that applies to their site. When properly developed, the EALs will be unambiguous, expressed in site-specific nomenclature and values, and be readily discernible from Control Room indications. If an example EAL does not apply because the intent cannot be met (e.g., specified instrumentation is not available at the plant), the developer should attempt to specify other available means for identifying entry into the IC.

- **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 need not be included in the site's emergency classification scheme basis document. In some cases, it may be appropriate to include information resulting from a developer note in the basis section.

- **ECL Assignment Attributes –** This sub-category of the Developer Notes provides a basis as to why the IC was assigned to a particular ECL. This information may or may not be included in the site-specific emergency classification scheme basis document.

3.6 IC AND EAL MODE APPLICABILITY

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**	**D**	**E**	**F**	**H**	**S**
Power Operations	X				X	X	X
Startup	X				X	X	X
Hot Standby	X				X	X	X
Hot Shutdown	X				X	X	X
Cold Shutdown	X	X				X	
Refueling	X	X				X	
Defueled (see below)	X	X				X	
None			X	X			

Typical BWR Operating Modes

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

Typical PWR Operating Modes

Power Operations (1):	Reactor Power > 5%, Keff ≥ 0.99
Startup (2):	Reactor Power ≤ 5%, Keff ≥ 0.99
Hot Standby (3):	RCS ≥ 350 °F, Keff < 0.99
Hot Shutdown (4):	200 °F < RCS < 350 °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

Developers will need to incorporate the mode criteria from unit-specific Technical Specifications into their emergency classification scheme. In addition, the scheme must also include the following mode designation specific to NEI 99-01:

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 99-01 to prepare a site-specific emergency classification scheme and the associated basis document.

It is important that the NEI 99-01 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

As indicated above, the guidance in NEI 99-01 is not intended to be applied to plants "as-is"; however, developers should attempt to keep their site-specific schemes as close to the guidance as possible. This 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.

As discussed in Section 3, the generic guidance includes Initiating Conditions (ICs) and example Emergency Action Levels (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 were not incorporated (i.e., sufficient detail to allow a third party to understand the decision not to incorporate the generic guidance).

For sites with more than one unit, consideration must be given to how events or conditions involving shared safety systems may affect more than one unit, and whether or not this should be a factor in an EAL that escalates the emergency classification level.

Useful acronyms and abbreviations associated with the NEI 99-01 emergency classification scheme are presented in Appendix C, Acronyms and Abbreviations.

Many words or terms used in the NEI 99-01 emergency classification scheme have scheme-specific definitions. These words and terms are identified by being set in all capital letters (i.e., ALL CAPS). The definitions are presented in Appendix D, Definitions.

Below are examples of acceptable changes to the generic guidance. These may be incorporated depending upon site 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 (e.g., IC SS7 may be renumbered to IC SS4).
- The first letter of a Recognition Category designation may be changed provided the change is carried through for all of the associated IC identifiers. For example, the Abnormal Radiation Levels / Radiological Effluent category designator "A" (for Abnormal) could 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 terms EAL and threshold may be used interchangeably.

4.2 PRESENTATION OF SCHEME INFORMATION FOR 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.

- Senior Reactor Operators in the Control Room are the first users of an emergency classification procedure. They may have other time-critical responsibilities during the emergency classification process, and may have little or no assistance in interpreting the ICs and EALs.
- 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 the licensed operators are comfortable. Developers must 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, alternate presentations of emergency classification scheme information may be developed for use by Emergency Directors and/or Offsite Response Organization personnel.

As an example of different approaches, a presentation method may involve the use of wall-boards. Two boards might be developed - one with information for power operations, startup and hot conditions, and the other for cold shutdown and refueling conditions. Alternative presentation methods for the Recognition Category F ICs, EALs and thresholds include flow charts, block diagrams, and checklist tables; the developer must ensure that the site-specific alternate method addresses all possible EAL and threshold combinations shown in the Recognition Category F Initiating Condition Matrix.

When providing EALs and user aids, such as wallboards, notes should be kept with each applicable EAL or moved to a common area and referenced by the applicable EAL. The expectation is that notes and other information necessary to classify the event will be on the wallboard, or other site-specific EAL presentation method, so that EAL decision-makers have this information readily available. It is not expected that similar notes be incorporated on EAL wallboards for every EAL; a reference to a Note on the EAL wallboard is acceptable as long as the information is adequately captured on the wallboard and pointed to for each applicable EAL.

4.3 LEVEL OF 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.

Developers may wish to consider placing visual cues (e.g., a step, note, caution, etc.) in plant procedures alerting the reader/user that it is appropriate 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.

4.4 BASIS DOCUMENT

A nuclear power plant will be required to prepare and submit an emergency classification scheme basis document as part of the NRC approval process for implementing an NEI 99-01 scheme. This basis document should list each Initiating Condition along with the applicable modes, associated EALs and/or thresholds and the supporting basis. A listing of defined terms and acronyms, and any necessary background or technical appendices, should also be included.

This document has several other useful purposes such as serving as a reference source when making an emergency classification assessment, providing information useful in training, supporting controls for configuration management and explaining an emergency classification to offsite authorities.

4.5 DEVELOPER AND USER FEEDBACK TO NEI

Questions or comments concerning the material presented in NEI 99-01 should be forwarded to the NEI EP Department staff. Staff members may provide a direct response (e.g., additional clarification or guidance), refer the feedback to the NEI 99-01 task force for a recommendation, coordinate resolution of issues with generic or industry-wide implications and/or take other action as deemed appropriate.

5 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

All IC and EAL assessments, and resulting classifications are to be based upon VALID indications, reports or conditions. See Appendix D for the definition of VALID.

For ICs and/or 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.

Events or conditions associated with planned evolutions to test, manipulate, repair, or perform maintenance or modifications to systems and equipment that result in an EAL being met or exceeded do not warrant an emergency classification provided that the evolution proceeds as planned, and the plant remains within the limits imposed by technical specifications or approved plans. These events or conditions may be subject to the reporting requirements of 10 CFR § 50.72.

5.1 CLASSIFICATION METHODOLOGY

The EALs specify the pre-determined, site-specific, observable thresholds for an IC that place the plant in a given Emergency Classification Level (ECL). 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. If it has, then the IC is considered met and the associated ECL is declared in accordance with plant procedures.

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 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 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.

- If an Alert EAL and a Site Area Emergency EAL are met, declare a Site Area Emergency.

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

- If two Alert EALs are met, declare an Alert.

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*.

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

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

5.4 CLASSIFICATION OF IMMINENT CONDITIONS

Although EALs provide specific thresholds, the Emergency Director must remain alert to events or conditions that could lead to the conclusion that exceeding an EAL 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 exceeded. 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.

5.5 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND DOWNGRADING

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

Emergency classification level upgrading for multi-unit stations with shared SAFETY-RELATED systems and functions must also consider the effects of a loss of a common system on more than one unit. For example, a two-unit station may have control panels for both units in close proximity to one another within the same room. Thus, an event requiring Control Room evacuation would most likely affect both units. There are a number of other systems and functions that may be shared at any given multi-unit station. This must be considered in the emergency classification level assessment.

An ECL may be downgraded when site-specific emergency classification 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	Ending Action
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.

5.6 CLASSIFYING TRANSIENT EVENTS AND 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 classification is warranted. In cases where no time-based criterion is specified, it is recognized that some transient events 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 events.

<u>EAL momentarily met during expected plant response</u> - In instances where an EAL is briefly met during an expected plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and relevant operator actions are appropriate.

<u>EAL momentarily met but the event or condition clears, or is corrected, prior to emergency declaration</u> - The key consideration is to determine if any plant damage occurred as a result of the event or condition.

- If the condition caused no plant damage and no further damage assessment is necessary, then the applicable EAL is not considered met and the associated emergency declaration is not required; however, the guidance contained in NUREG-1022, Section 3.1.1 is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 within one hour, and notification of State and local response organizations should be made in accordance with the arrangements made between the site and offsite response organizations.
- If the condition caused plant damage, or if damage assessment is necessary to confirm or rule-out such damage, then the applicable EAL should be considered met and the appropriate emergency declaration made. In cases where an assessment is required, terminate the emergency in accordance with plant procedures after the assessment is complete and other termination criteria are met.

<u>EAL met but the emergency classification was not made at the time of the event or condition</u> - This situation occurs 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, 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, and notification made to State and local emergency response organizations accordance with the arrangements established between the licensee and offsite organizations.

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 Any release of gaseous or liquid radioactivity to the environment greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer. *Op.* Modes*: All*	**AA1** Any release of gaseous or liquid radioactivity to the environment greater than 200 times the (site-specific effluent release controlling document) limits for 15 minutes or longer. *Op.* Modes*: All*	**AS1** Actual or projected offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE. *Op.* Modes*: All*	**AG1** Actual or projected offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE. *Op. Modes: All*
AU2 UNPLANNED loss of water level covering irradiated fuel. *Op. Modes: All*	**AA2** Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel. *Op.* Modes*: All*		
AU3 UNPLANNED rise in plant radiation levels. *Op. Modes: All*			

AU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Any release of gaseous or liquid radioactivity to the environment greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.

Operating Mode Applicability: All

Example Emergency Action Levels:

Note: In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

(1) Reading on **ANY** effluent radiation monitor greater than (2 times the site-specific effluent controlling document limits) for 60 minutes or longer:

(site-specific monitor list and threshold values corresponding to 2 times the controlling document 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) Confirmed sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times (site-specific effluent release controlling document limits) for 60 minutes or longer.

(4) Reading on perimeter radiation monitoring system reading greater than 0.10 mR/hr above NORMAL LEVELS for 60 minutes or longer. [for sites having telemetered perimeter monitors capable of reading this value]

(5) Indication on automatic real-time dose assessment capability indicating greater than (site-specific value) for 60 minutes or longer. [for sites having such capability]

Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release).

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.

The (site-specific effluent controlling document limit) multiples are specified in AU1 and AA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

This IC includes any radiological release, gaseous or liquid, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the associated release path to the environment has been isolated, the effluent monitor reading is no longer VALID.

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 normally occurring continuous radioactivity releases from 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 is 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.).

EALs #4 and #5

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

EAL #1 and #2 use monitor readings/setpoints that were calculated using an assumed annual average meteorology. Values assessed against EALs #4 and #5 will be a function of actual meteorology, which will be different than the assumed annual average. Thus, there will be numerical differences.

Developer Notes:

ECL Assignment Attributes: 3.1.1.B and 3.1.1.D

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

Radiological release limits are typically located in the Offsite Dose Calculation Manual (ODCM) or for plants that have not implemented Generic Letter 89-01[2] in the Radiological Effluent Technical Specifications (RETS).

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 exceeding 2 times a release control limit. The release control documents typically describe methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values.

For EAL #2 - Values in this EAL should be 2 times the setpoint established by the release control document to warn of a release that is not in compliance with the specified limits. Indexing the EAL to the control document setpoints in this manner ensures that the EAL will not be less than the setpoint established by a specific discharge permit.

Developers should research historical discharge permit setpoints to ensure the applicable monitor's operating range will indicate the highest reasonably expected EAL value (2 times the setpoint). If necessary, establish an EAL value appropriate for conditions when the monitor may be over-ranged by the 2 times setpoint value.

For EALs #4 and #5 - As provided in the effluent release control document, this value is determined by prorating the annual allowed dose over 8,760 hours, multiplying by 2 and rounding down.

365 ~~days~~/year * 24 hours/~~day~~ = 8,760 hours/year

(500 mrem/~~year~~ ÷ 8,760 hours/~~year~~) = 0.057 mrem/hour

0.057 mrem/hour * 2 = .114 mrem/hour or 0.10 mrem/hour

[2] *Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program*

AU2

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of water level covering irradiated fuel.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) UNPLANNED water level drop in (site-specific reactor refueling pathway) as indicated by (site-specific radiation monitor indication or survey).

Basis:

This IC addresses a decrease in water level above irradiated (spent) 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.

The refueling pathway is (the site-specific combination of cavities, tubes, canals and pools).

A significant drop in the water level above irradiated fuel will cause an increase in adjacent area radiation levels. Increases in area radiation levels may be detected by radiation monitors and/or surveys.

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 water level drop”.

For refueling outage events where the water level drops below the reactor vessel flange, classification would be via IC CU2. This event escalates to an Alert per AA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table, when applicable.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.D

Site-specific indications may include instrumentation values such as water level and area radiation monitor readings. Reports from personnel may also be included (e.g., from a refueling crew). If available, video cameras may allow for remote observation. Depending on available instrumentation, the declaration may also be based on indications of water makeup rate and/or decreases in the level of a water storage tank.

AU3

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

UNPLANNED rise in plant radiation levels.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) UNPLANNED rise in area radiation monitor readings or survey results by a factor of 1,000 over NORMAL LEVELS.

Basis:

This IC addresses significantly elevated plant radiation levels caused by an UNPLANNED event or condition. The magnitude of the increase is indicative of a minor loss in the ability to control radiation levels within the plant or radioactive materials. This condition is therefore a potential degradation in the level of safety of the plant.

The EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials.

It is recognized that some plant area radiation monitors may not be able to detect or display a reading that is 1,000 times NORMAL LEVELS. The intent of this IC is to rely on currently installed plant monitors and not to require design changes/backfits. In cases where an installed area radiation monitor cannot detect or display values at or above 1,000 times NORMAL LEVELS value, then survey results may be used. It is also acceptable to estimate a value through extrapolation (e.g., using a reading from a nearby on-range monitor or a survey at a distance from the area).

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.D

Developers should ensure that any list of site-specific area radiation monitors (if provided) is not unduly restrictive to the evaluation of the IC and EAL. The intent is to identify loss of control of radioactive material in any monitored area.

AA1

Initiating Condition - ALERT

Any release of gaseous or liquid radioactivity to the environment greater than 200 times the (site-specific effluent release controlling document) limits for 15 minutes or longer.

Operating Mode Applicability: All

Example Emergency Action Levels:

Note: In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

(1) Reading on **ANY** effluent radiation monitor greater than (200 times the site-specific effluent controlling document limits) for 15 minutes or longer:

(site-specific monitor list and threshold values corresponding to 200 times the controlling document limits)

(2) Reading on **ANY** effluent radiation monitor greater than 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

(3) Confirmed sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 200 times (site-specific effluent release controlling document limits) for 15 minutes or longer.

(4) Reading on perimeter radiation monitoring system reading greater than 10.0 mR/hr above NORMAL LEVELS for 15 minutes or longer. [for sites having telemetered perimeter monitors capable of reading this value]

(5) Indication on automatic real-time dose assessment capability indicating greater than (site-specific value) for 15 minutes or longer. [for sites having such capability]

Basis:

This IC addresses an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory commitments for an extended period of time (e.g., a significant uncontrolled release).

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.

The (site-specific effluent controlling document limit) multiples are specified in AU1 and AA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

This IC includes any radiological release, gaseous or liquid, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the associated release path to the environment has been isolated, the effluent monitor reading is no longer VALID.

Releases should not be prorated or averaged. For example, a release exceeding 600 times release limits for 5 minutes does not meet the EAL.

<u>EAL #1</u>

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

<u>EAL #2</u>

This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 200 times the limit established by a radioactivity discharge permit. This EAL is associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

<u>EAL #3</u>

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.).

<u>EALs #4 and #5</u>

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

EAL #1 and #2 use monitor readings/setpoints that were calculated using an assumed annual average meteorology. Values assessed against EALs #4 and #5 will be a function of actual meteorology, which will be different than the assumed annual average. Thus, there will be numerical differences.

Developer Notes:

ECL Assignment Attributes: 3.1.2.C

Refer to Appendix A for the detailed basis of the radiological effluent IC/EALs.

Radiological release limits are typically described in the Offsite Dose Calculation Manual (ODCM), or for plants that have not implemented Generic Letter 89-01 in the Radiological Effluent Technical Specifications (RETS).

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 exceeding 200 times a release control limit. The release control documents typically describe methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values.

For EAL #2 – Values in this EAL should be 200 times the setpoint established by the effluent release control document to warn of a release that is not in compliance with the specified limits. Indexing the EAL to the control document setpoints in this manner ensures that the EAL will not be less than the setpoint established by a specific discharge permit.

Developers should research historical discharge permit setpoints to ensure the applicable monitor's operating range will indicate the highest reasonably expected EAL value (200 times the setpoint). If necessary, establish an EAL value appropriate for conditions when the monitor may be over-ranged by the 200 times value.

For EAL #4 and #5 - As provided in the effluent release control document, this value is determined by prorating the annual allowed dose over 8,760 hours, multiplying by 200 and rounding down.

365 ~~days~~/year * 24 hours/~~day~~ = 8,760 hours/year

(500 mrem/~~year~~ ÷ 8,760 hours/~~year~~) = 0.057 mrem/hour

0.057 mrem/hour * 200 = 11.4 mrem/hour or 10 mrem/hour

AA2

Initiating Condition - ALERT

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) A water level drop in (site-specific reactor refueling pathway) that will result in irradiated fuel becoming uncovered.

(2) Damage to irradiated fuel or loss of water level as indicated by (site-specific alarm and/or elevated reading) on **ANY** of the following:

(site-specific radiation monitors)

Basis:

This IC addresses events that may lead to, or conditions that are indicative of, damage to irradiated (spent) fuel assemblies outside of the reactor vessel. These events present safety challenges to plant personnel and may be 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.

<u>EAL #1</u>

The refueling pathway is (the site-specific combination of cavities, tubes, canals and pools).

<u>EAL #2</u>

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

While an area radiation monitor could detect an increase in dose rate due to a drop in the water level, the reading may not be a reliable indication of whether or not the fuel is covered. Increased ventilation monitor readings may be an indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background radiation at the ventilation monitor due to a water level decrease (i.e., “shine”) may affect ventilation monitor readings and needs to be considered.

Escalation of this emergency classification level, if appropriate, would be based on ICs AS1 or AG1.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B and 3.1.2.C

These events escalate from AU2 in that a release of radioactivity from irradiated fuel is anticipated or has occurred. This IC applies to irradiated (spent) fuel requiring water cooling and is not intended to apply to irradiated fuel which is licensed for dry storage.

For EAL #1 - Site-specific indications may include instrumentation values such as water level and area radiation monitor readings, and personnel reports (e.g., from a refueling crew). If available, video cameras may allow for remote observation. Depending on available instrumentation, the declaration may also be based on indications of water makeup rate and/or decreases in the level of a water storage tank.

For EAL #2 – Developers need to ensure that a specific list of area radiation monitors is not unduly restrictive to the evaluation of the IC and EAL. The intent is to identify loss of control of radioactive material in any monitored area.

AS1

Initiating Condition -- SITE AREA EMERGENCY

Actual or projected offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary.

(2) Field survey results indicate **EITHER** of the following at or beyond the site boundary:

- 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.

(3) Perimeter radiation monitoring system reading greater than 100 mR/hr for 15 minutes or longer. [for sites having telemetered perimeter monitors capable of reading this value]

Basis:

This IC addresses a release of radioactivity that results in projected or actual doses at or beyond the site boundary greater than or equal to 10% of the EPA PAGs. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. To ensure a timely assessment of this IC, emergency plan implementing procedures should prompt timely performance of dose assessments using actual meteorology.

Developer Notes:

ECL Assignment Attributes: 3.1.3.C

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

While this IC could not realistically occur absent challenges to multiple fission product barriers, the IC provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status alone.

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

The TEDE dose is set at 10% of the EPA PAG 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.

Unlike IC AU1 and AA1, AS1 does not have an EAL based on a calculated effluent monitor reading using assumed meteorology for the following reasons.

- This IC is bound by the Fission Product Barrier ICs which would realistically result in the same or higher classification prior to reaching the pre-determined dose and dose rate values.

- Dose projection results using an assumed meteorology will differ significantly from those using actual meteorology. A declaration of a Site Area Emergency may result in the implementation of offsite precautionary or protective actions; these actions involve some cost and risk to the public, and must be based on accurate, real-time dose projections.

- Since dose assessment capability is required to be available, a default assumed meteorology EAL would be bound by the dose assessment EAL which uses actual meteorological data.

- Calculated default effluent monitor thresholds for the Site Area Emergency and General Emergency levels may require use of an emergency dose assessment methodology to adequately determine EAL values. This methodology is significantly from that used to assess routine effluent releases, and which provides the basis for the Unusual Event and the Alert EALs. The differing methodologies can lead to overlapping, or insufficiently separated, Alert and Site Area Emergency dose and dose rate thresholds.

- The Alert and Site Area Emergency EALs have different underlying bases. The Alert EALs are based on 200 times an ODCM/RETS limit while the Site Area Emergency EALs are based 10% of the EPA PAG limits. The differing bases can lead to overlapping, or insufficiently separated, Alert and Site Area Emergency dose and dose rate thresholds.

AG1

Initiating Condition -- GENERAL EMERGENCY

Actual or projected offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the site boundary.

(2) Field survey results indicate **EITHER** of the following at or beyond the site boundary:

- 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, at or beyond site boundary.

(3) Perimeter radiation monitoring system reading greater than 1,000 mR/hr for 15 minutes or longer. [for sites having telemetered perimeter monitors capable of reading this value]

Basis:

This IC addresses a release of radioactivity that results in projected or actual doses at or beyond the site boundary greater than or equal to the EPA Protective Action Guides (PAGs). Releases of this magnitude will require implementation of protective actions for the public. To ensure a timely assessment of this IC, emergency plan implementing procedures should prompt timely performance of dose assessments using actual meteorology.

Developer Notes:

ECL Assignment Attributes: 3.1.4.C

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

While this IC could not realistically occur absent challenges to multiple fission product barriers, the IC provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status alone. For more severe accident sequences, there may be large uncertainties associated with the source term and/or the release may be unmonitored.

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

The TEDE dose is set at 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.

Unlike IC AU1 and AA1, AG1 does not have an EAL based on a calculated effluent monitor reading using assumed meteorology for the following reasons.

- This IC is bound by the Fission Product Barrier ICs which would realistically result in the same or higher classification prior to reaching the pre-determined dose and dose rate values.

- Dose projection results using an assumed meteorology will differ significantly from those using actual meteorology. A declaration of a General Emergency will result in the implementation of offsite protective actions; these actions involve some cost and risk to the public, and must be based on accurate, real-time dose projections.

- Since dose assessment capability is required to be available, a default assumed meteorology EAL would be bound by the dose assessment EAL which uses actual meteorological data.

- Calculated default effluent monitor thresholds for the Site Area Emergency and General Emergency levels may require use of an emergency dose assessment methodology to adequately determine EAL values. This methodology is significantly from that used to assess routine effluent releases, and which provides the basis for the Unusual Event and the Alert EALs. The differing methodologies can lead to overlapping, or insufficiently separated, Alert and Site Area Emergency dose and dose rate thresholds.

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 RCS leakage for 15 minutes or longer. *Op. Modes: Cold Shutdown*	**CA1** Loss of RCS/reactor vessel inventory. *Op. Modes: Cold Shutdown, Refueling*	**CS1** Loss of RCS/reactor vessel inventory affecting core decay heat removal capability. *Op. Modes: Cold Shutdown, Refueling*	**CG1** Loss of RCS/reactor vessel inventory affecting fuel clad integrity with containment challenged. *Op. Modes: Cold Shutdown, Refueling*
CU2 RCS leakage for 15 minutes or longer. *Op. Modes: Refueling*			
CU3 AC power capability to emergency busses reduced to a single power source for 15 minutes or longer. *Op. Modes: Cold Shutdown, Refueling, Defueled*	**CA3** Loss of all offsite and all onsite AC power to emergency busses for 15 minutes or longer. *Op. Modes: Cold Shutdown, Refueling, Defueled*		
CU4 UNPLANNED loss of decay heat removal capability. *Op.* Modes*: Cold Shutdown, Refueling*	**CA4** Inability to maintain the plant in cold shutdown. *Op. Modes: Cold Shutdown, Refueling*		
CU5 Loss of required DC power for 15 minutes or longer. *Op. Modes: Cold Shutdown, Refueling*			
CU6 Inadvertent criticality. *Op. Modes: Cold Shutdown, Refueling*			

CU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

RCS leakage for 15 minutes or longer.

Operating Mode Applicability: Cold Shutdown

Example Emergency Action Levels:

(1) RCS leakage results in the inability to restore and maintain RPV level greater than (site-specific low level RPS actuation setpoint) for 15 minutes or longer. [*BWR*]

(1) RCS leakage results in the inability to restore and maintain (site-specific pressurizer or reactor vessel minimum level or target bands) for 15 minutes or longer. [*PWR*]

Basis:

This IC addresses the inability to restore and maintain water level at a required level/band, and is indicative of a loss of RCS inventory. This is considered to be a potential degradation of the level of safety of the plant. ICs CU1 and CU2 are included to reflect the different RCS conditions that exist in the cold shutdown and refueling modes.

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

If the EAL is met due to the as-designed/expected operation of a relief valve, no emergency classification is warranted. An emergency classification would be required if the RCS leakage 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).

Continued loss of RCS Inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA4.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

CU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

RCS leakage for 15 minutes or longer.

Operating Mode Applicability: Refueling

Example Emergency Action Levels:

(1) a. RCS/reactor vessel level band is established at or above the reactor vessel flange.

AND

b. UNPLANNED RCS/reactor vessel water level drop below the reactor vessel flange for 15 minutes or longer.

(2) a. RCS/reactor vessel level band is established below reactor vessel flange.

AND

b. UNPLANNED RCS/reactor vessel water level drop below the established RCS/reactor vessel level band for 15 minutes or longer.

(3) a. RCS/reactor vessel level cannot be monitored.

AND

b. UNPLANNED level rise in (site-specific sump and/or tank) due to a loss of RCS/reactor vessel inventory.

Basis:

This IC addresses the inability to restore and maintain water level at required levels, or a loss of RCS/reactor vessel level indications. Either of these conditions is considered to be a potential degradation of the level of safety of the plant. ICs CU1 and CU2 are included to reflect the different RCS conditions that exist in the cold shutdown and refueling modes.

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

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

EAL #3 addresses conditions in the refueling mode when the normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication are normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. If all level indication were lost during a loss of RCS inventory event, operators would need to identify the inventory loss by observing changes to various sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

This IC is not applicable to decreases in a “flooded” reactor cavity. This condition is addressed by IC AU2, EAL1, until such time as the level decreases to the level of the vessel flange.

Continued loss of RCS inventory will result in escalation to the Alert emergency classification level via either IC CA1 or CA4.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

CU3

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

AC power capability to emergency busses reduced to a single power source for 15 minutes or longer.

Operating Mode Applicability: Cold Shutdown, Refueling, Defueled

Example Emergency Action Levels:

(1) AC power capability to (site-specific emergency busses) is reduced to a single power source for 15 minutes or longer.

Basis:

This IC describes a significant degradation of AC power sources (offsite and onsite) such that any additional single failure would result in a loss of all AC emergency busses.

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

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA3.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B

Some potential examples of this condition are presented below.

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

Developers should consider including site-specific examples in their IC CU3 basis.

At multi-unit stations, the EALs should allow credit for compensatory measures that 1) are proceduralized, and 2) can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, “swing” generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to cross-tie AC power from an offsite power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC. These stations must also consider the impact of this condition on SAFETY-RELATED functions shared between multiple units.

CU4

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of decay heat removal capability.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

(1) UNPLANNED loss of decay heat removal results in RCS temperature greater than the Technical Specification cold shutdown temperature limit.

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

Basis:

An UNPLANNED loss of decay heat removal capability is a potential degradation of the level of safety of the plant.

During refueling, 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 rise in reactor coolant temperature depending on the time after shutdown.

EAL #1 reflects a condition where decay heat removal capability has been degraded or lost 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. The core decay heat load has been reduced since the cessation of power operation and there is a large volume of reactor coolant available to act as a heat sink.

EAL #2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions; operators would be unable to monitor key parameters necessary to assure core decay heat removal. Again, the core decay heat load has been reduced since the cessation of power operation and there is a large volume of reactor coolant available to act as a heat sink. There is no immediate threat of fuel damage.

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 CA4 based on exceeding configuration-specific time criteria.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B

CU5

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of required DC power for 15 minutes or longer.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

(1) Indicated voltage is less than (site-specific bus voltage indication) on (required site-specific Vital DC busses) for 15 minutes or longer.

Basis:

The purpose of this IC and its associated EAL is to recognize a loss of DC power compromising the ability to monitor and control plant systems. This condition is considered to be a potential degradation of the level of safety of the plant.

As used in this IC and EAL, "required" means the DC busses necessary to support operations of the in-service train or trains of SAFETY-RELATED equipment. For example, if the Train A ECCS is out-of-service for scheduled maintenance work and the Train B ECCS is in-service (operable and ready for use), then the "required" DC busses are those necessary to support operation of the B Train ECCS equipment. A loss of the Train A DC busses would not warrant an emergency classification.

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

If this loss results in the inability to maintain cold shutdown, the escalation to an Alert would be in accordance with IC CA4.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B

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

CU6

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inadvertent criticality.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

(1) An UNPLANNED sustained positive period observed on nuclear instrumentation. [*BWR*]

(1) An UNPLANNED sustained positive startup rate observed on nuclear instrumentation. [*PWR*]

Basis:

This IC addresses an inadvertent criticality event that occurs in the Cold Shutdown or Refueling mode. Such events may result from improper fuel loading or an unplanned dilution. This IC indicates a potential degradation of the level of safety of the plant and warrants a NOUE classification.

This condition can be identified using period monitors/startup rate monitors. The term “sustained” is used in order to allow exclusion of expected short-term positive periods/startup rates from planned fuel bundle or control rod movements during core alterations. These short term positive periods/startup rates are the result of the increase in neutron population due to subcritical multiplication.

Escalation would be by Emergency Director judgment.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

Refer to NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*, for additional information.

CA1

Initiating Condition - ALERT

Loss of RCS/reactor vessel inventory.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

(1) Loss of RCS/reactor vessel inventory as indicated by level less than (site-specific level).

(2) a. RCS/reactor vessel level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED level rise in (site-specific sump and/or tank) due to a loss of RCS/reactor vessel inventory.

Basis:

The conditions described by these EALs 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 warrants an Alert declaration.

EAL #1

A lowering of water level below (site-specific EAL point) indicates that operator actions have not been successful in restoring RCS/reactor vessel level. A continuing decrease in RCS/reactor vessel level will lead to core uncovery.

EAL #2

If all level indication were unavailable during a loss of RCS/reactor vessel inventory event, operators would need to determine that the inventory loss was occurring by observing changes to sump and/or tank levels. Sump and/or tank level increases must be evaluated against other potential sources of leakage to ensure they are indicative of leakage from the RCS/reactor vessel.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CG1 basis.

If the RCS/reactor vessel level continues to lower, then escalation to Site Area Emergency will be via IC CS1.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

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

- Low-Low ECCS actuation setpoint / Level 2 [BWR]
- Bottom Inside Diameter (ID) of the RCS loop [PWR]

The BWR Low-Low ECCS Actuation Setpoint/Level 2 was chosen because it is a standard operationally significant setpoint at which some (typically high pressure ECCS) injection systems would automatically start and is a value significantly below the low RPV water level RPS actuation setpoint specified in IC CU1.

The PWR Bottom ID of the RCS Loop Setpoint was chosen because a loss of the reactor coolant recirculation suction point in the loop is imminent or has occurred, and RCS/reactor vessel level indication may off-scale low or nearly so. The Bottom ID of the RCS loop setpoint should be the level equal to the bottom of the reactor vessel loop penetration (and not the low point of the loop).

For EAL #2 - In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

CA3

Initiating Condition - ALERT

Loss of all offsite and all onsite AC power to emergency busses for 15 minutes or longer.

Operating Mode Applicability: Cold Shutdown, Refueling, Defueled

Example Emergency Action Levels:

(1) Loss of **ALL** offsite and **ALL** onsite AC Power to (site-specific emergency busses) for 15 minutes or longer.

Basis:

This event involves an actual or potential substantial degradation of the level of safety of the plant, and warrants an Alert declaration. A loss of all AC power compromises all plant SAFETY-RELATED systems requiring electric power including systems required for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and ultimate heat sink. This type of event is classified as an Alert (and not a Site Area Emergency) when in the cold shutdown, refueling, or defueled mode because of the significantly reduced core decay heat load, lower temperature and pressure in various systems, and the increased time available to restore one of the emergency busses to service.

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

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

At multi-unit stations, the EALs should allow credit for compensatory measures that 1) are proceduralized and 2) can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, “swing” generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to cross-tie AC power from an offsite power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC. These stations must also consider the impact of this condition on SAFETY-RELATED functions shared between multiple units.

CA4

Initiating Condition - ALERT

Inability to maintain the plant in cold shutdown.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

(1) UNPLANNED loss of decay heat removal capability resulting in RCS temperature greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in (site-specific table).

Table: RCS Reheat Duration Threshold		
CNMT Status / RCS Status	CONTAINMENT CLOSURE not established	CONTAINMENT CLOSURE established
RCS Not Intact	0 minutes	20 minutes*
RCS at reduced inventory (mid-loop operation) [*PWR*]	0 minutes	20 minutes*
RCS Intact	60 minutes*	60 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) UNPLANNED loss of decay heat removal capability resulting in an RCS pressure increase greater than 10 psi. (PWR note - This EAL does not apply during water-solid plant conditions.)

Basis:

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

The RCS Reheat Duration Threshold table addresses a complete loss of core cooling functions when CONTAINMENT CLOSURE is established but RCS integrity is not established (not intact), or RCS inventory is reduced (e.g., mid-loop operation in PWRs). The 20-minute criterion was included to allow time for operator action to restore the heat removal function, if possible.

The RCS Reheat Duration Threshold table also addresses a complete loss of core cooling functions with RCS integrity established (intact). The status of the containment barrier or 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 restore cooling without there being a substantial degradation in plant safety.

In the case where there is a complete loss of core cooling functions, and when the RCS is not intact or is at reduced inventory, and the containment barrier is not functional, no reheat duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the Containment atmosphere, and 2) there is reduced reactor coolant inventory above the top of active fuel (TOAF). Even though fuel damage is not immediate, two fission product barriers, the RCS and containment, are not available.

EAL #2 provides an alternate indication of an UNPLANNED RCS heatup beyond (site-specific Technical Specification cold shutdown temperature limit). This alternate indication may be used during Cold Shutdown, with the RCS intact, in the event that all RCS temperature indication is unavailable.

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

Escalation to Site Area Emergency would be via CS1 should boiling result in significant reactor vessel level loss.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

For EAL #1 - RCS should be considered intact or not intact in accordance with site-specific criteria.

For EAL #2 - The RCS pressure setpoint chosen should be 10 psi or the lowest change in pressure increment that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psi.

For PWRs, 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.

CS1

Initiating Condition - SITE AREA EMERGENCY

Loss of RCS/reactor vessel inventory affecting core decay heat removal capability.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

(1) a. CONTAINMENT CLOSURE not established.

AND

b. RCS/reactor vessel level less than (site-specific level).

(2) a. CONTAINMENT CLOSURE established.

AND

b. RCS/reactor vessel level less than (site-specific level).

(3) a. RCS/reactor vessel level cannot be monitored.

AND

b. Core uncovery is indicated by **ANY** of the following for 30 minutes or longer:

- (Site-specific radiation monitor) reading greater than (site-specific value)
- Erratic source range monitor indication [*PWR*]
- (Site-specific UNPLANNED changes sump and/or tank levels of sufficient magnitude to indicate core uncovery)
- (Other site-specific indications)

Basis:

This IC addresses a significant and prolonged loss of RCS/reactor vessel inventory control and makeup leading to IMMINENT fuel damage. The lost inventory may be due to an RCS breach, loss of configuration control or prolonged boiling of reactor coolant. These conditions are consistent with the attributes of a Site Area Emergency.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs 1.b and 2.b reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

Following an extended loss of RCS/reactor vessel heat removal and makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS/reactor vessel level cannot be restored, fuel damage is probable. Analyses have indicated that core damage may occur within an hour following continued core uncovery; therefore, 30 minutes is used to provide some margin for various uncertainties.

Escalation to a General Emergency is via Initiating Conditions CG1 or AG1.

EAL #3

The 30-minute duration allows sufficient time for actions to be performed to terminate the leakage and/or recover inventory control/makeup equipment.

If all level indication were unavailable during a loss of RCS/reactor vessel inventory event, operators would need to determine that the inventory loss was occurring by observing sump and/or tank level changes. Sump and/or tank level increases must be evaluated against other potential sources of leakage to ensure they are indicative of leakage from the RCS/reactor vessel.

Developer Notes:

ECL Assignment Attributes: 3.1.3.B

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

PWR

For EAL #1.b - site-specific level at 6" below the bottom ID of the RCS loop. This is the level at 6” below the bottom ID of the reactor vessel penetration and not the low point of the loop. PWRs unable to measure this level should choose the first observable level below the bottom ID of the loop as the EAL value. If the range of water level instrumentation is such that the EAL value cannot be evaluated, then EAL 3 should be used to determine if the IC has been met.

For EAL #2.b – Insert level indication corresponding to top of active fuel.

For EAL #3.b - As water level in the reactor vessel lowers, the dose rate above the core will increase. A site-specific dose rate value indicative of core uncovery (i.e., level at TOAF) should be estimated and included as an EAL.

For EAL #3.b - 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.

For EAL #3.b – A threshold based on changes to sump and/or tank levels should use estimated values indicative of enough water loss to potentially uncover the core. It is recognized that some plant designs may not support development of a useful and reliable indication for this threshold.

BWR

For EAL #1 - site-specific level at the Low-Low-Low ECCS actuation setpoint / Level 1. The BWR Low-Low-Low ECCS actuation setpoint / Level 1 was chosen because it is a standard operationally significant setpoint at which some (typically low pressure ECCS) injection systems would automatically start and attempt to restore RPV level. This is a RPV water level value that is observable below the Low-Low/Level 2 value specified in IC CA1, but significantly above the Top of Active Fuel (TOAF) threshold specified in EAL #2.

For EAL #1.b - Since BWRs have RCS penetrations below the EAL value, continued level decrease may be indicative of pressure boundary leakage.

For EAL #2.b – Insert level indication corresponding to top of active fuel.

For EAL #3.b - As water level in the reactor vessel lowers, the dose rate above the core will increase. A site-specific dose rate value indicative of core uncovery (i.e., level at TOAF) should be estimated and included as an EAL. For BWRs that do not have installed radiation monitors capable of indicating core uncovery, alternate site-specific level indications of core uncovery should be used if available.

For EAL #3.b - Because BWR source range monitor (SRM) nuclear instrumentation detectors are typically located below core mid-plane, this is not a viable indicator of core uncovery for BWRs.

For EAL #3.b – A threshold based on changes to sump and/or tank levels should use estimated values indicative of enough water loss to potentially uncover the core. It is recognized that some plant designs may not support development of a useful and reliable indication for this threshold.

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

CG1

Initiating Condition - GENERAL EMERGENCY

Loss of RCS/reactor vessel inventory affecting fuel clad integrity with containment challenged.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

(1) a. RCS/reactor vessel level less than (site-specific level) for 30 minutes or longer.

AND

b. **ANY** containment challenge indication (see Table):

(2) a. RCS/reactor vessel level cannot be monitored.

AND

b. Core uncovery is indicated by **ANY** of the following for 30 minutes or longer.

- (Site-specific radiation monitor) reading greater than (site-specific value).
- Erratic source range monitor indication [*PWR*].
- (Site-specific UNPLANNED changes sump and/or tank levels of sufficient magnitude to indicate core uncovery).
- (Other site-specific indications)

AND

c. **ANY** containment challenge indication (see Table):

Table: Containment Challenge Indications
■ CONTAINMENT CLOSURE not established.
■ Explosive mixture exists inside containment.
■ UNPLANNED rise in containment pressure.
■ Secondary containment radiation monitor reading above (site-specific value). [*BWR*]

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 RCS/reactor vessel heat removal and makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS/reactor vessel level cannot be restored, fuel damage is probable. Analyses have indicated that core damage may occur within an hour following continued core uncovery; therefore, 30 minutes is used to provide some margin for various uncertainties.

If all level indication were unavailable during a loss of RCS/reactor vessel inventory event, operators would need to determine that the inventory loss was occurring by observing sump and/or tank level changes. Sump and/or tank level increases must be evaluated against other potential sources of leakage to ensure they are indicative of leakage from the RCS/reactor vessel.

With the CONTAINMENT barrier open or challenged, 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 core uncovery time limit, then escalation to a General Emergency is not required.

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 containment challenge indications.

Developer Notes:

ECL Assignment Attributes: 3.1.4.B

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

For EAL #1.a – Insert level indication corresponding to top of active fuel.

For EAL #2.b - As water level in the reactor vessel lowers, the dose rate above the core will increase. A site-specific dose rate value indicative of core uncovery (i.e., level at TOAF) should be estimated and included as an EAL. For BWRs that do not have installed radiation monitors capable of indicating core uncovery, alternate site-specific level indications of core uncovery should be used if available.

For EAL #2.b - Post-TMI 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. Because BWR Source Range Monitor (SRM) nuclear instrumentation detectors are typically located below core mid-plane, this is not a viable indicator of core uncovery.

For EAL #2.b – A threshold based on changes to sump and/or tank levels should use estimated values indicative of enough water loss to potentially uncover the core. It is recognized that some plant designs may not support development of a useful and reliable indication for this threshold.

For the Containment Challenge Indications Table:

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

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

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

8 PERMANENTLY DEFUELED STATION ICS/EALS

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

UNUSUAL EVENT	ALERT
D-AU1 Any release of gaseous or liquid radioactivity to the environment greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer. Op. *Modes: Not Applicable*	**D-AA1** Any release of gaseous or liquid radioactivity to the environment greater than 200 times the (site-specific effluent release controlling document) limits for 15 minutes or longer. *Op. Modes: Not* Applicable
D-AU2 UNPLANNED rise in plant radiation levels. *Op. Modes: Not Applicable*	**D-AA2** Rise in radiation levels within the facility that impedes operation of systems required to maintain spent fuel integrity. *Op. Modes: Not Applica*ble
D-SU1 UNPLANNED spent fuel pool temperature rise. Op. *Modes: Not Applicable*	
D-HU1 Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant. *Op. Modes: Not Applicable*	**D-HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat. *Op. Modes: Not* Applicable
D-HU2 Natural or destructive phenomena inside the PROTECTED AREA affecting the ability to maintain spent fuel integrity. *Op.* Modes*: Not Applicable*	
D-HU3 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE. *Op. Modes: Not Applicable*	**D-HA3** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert. *Op. Modes: Not* Applicable

D-AU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Any release of gaseous or liquid radioactivity to the environment greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.

Operating Mode Applicability: Not Applicable

Example Emergency Action Levels:

Note: In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

(1) Reading on **ANY** effluent radiation monitor greater than (2 times the site-specific effluent controlling document limits) for 60 minutes or longer:

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

(2) Confirmed sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times (site-specific effluent release controlling document 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 radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release).

Defueled 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.

The (site-specific effluent controlling document limit) multiples are specified in D-AU1 and D-AA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the associated release path to the environment has been isolated, the effluent monitor reading is no longer VALID.

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 release pathways.

EAL #2

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.).

Developer Notes:

ECL Assignment Attributes: 3.1.1.D

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

Radiological release controls limits are typically located in the Off-site Dose Calculation Manual (ODCM), or for plants that have not implemented Generic Letter 89-01, in the Radiological Effluent Technical Specifications (RETS).

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 exceeding 2 times a release control limit. The release control documents typically describe methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values.

Calculations supporting the release rates specified in the EAL values should be provided which quantify expected doses at the Restricted Area Boundary. The major isotope of concern in the permanently defueled condition is Kr-85.

D-AU2

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

UNPLANNED rise in plant radiation levels.

Operating Mode Applicability: Not Applicable

Example Emergency Action Levels:

(1) a. UNPLANNED water level drop in the spent fuel pool as indicated by (site-specific level or indication).

AND

b. Area Radiation Monitor reading rise on (site-specific list).

(2) Area radiation monitor reading or survey result indicates an UNPLANNED rise of 25 mR/hr over NORMAL LEVELS.

Basis:

This IC addresses elevated plant radiation levels caused by a decrease in water level above irradiated (spent) fuel or other UNPLANNED events. The increased radiation levels are indicative of a minor loss in the ability to control radiation levels within the plant or radioactive materials. Either condition is a potential degradation in the level of safety of the plant.

EAL #2 excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials.

Developer Notes:

ECL Assignment Attributes: 3.1.1.D

For EAL #1 - Site-specific indications may include instrumentation values such as water level and area radiation monitor readings, and personnel reports. If available, video cameras may allow for remote observation. Depending on available instrumentation, the declaration may also be based on indications of water makeup rate and/or decreases in the level of a water storage tank.

For EAL #2 - The specified value of 25 mR/hr is arbitrary and may be set to another value for a specific application with appropriate justification.

D-SU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

UNPLANNED spent fuel pool temperature rise.

Operating Mode Applicability: Not Applicable

Example Emergency Action Levels:

(1) UNPLANNED spent fuel pool temperature rise greater than (site-specific °F).

Basis:

Classification of this condition as a NOUE is warranted since it is a precursor to more serious event and represents a potential degradation in the level of safety of the plant. If uncorrected, boiling in the pool will occur, and result in a loss of pool level and increased radiation levels.

In-plant dose rates will drive escalation of the emergency via IC D-AA2.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B.

The site-specific temperature should be chosen based on the starting point for fuel damage calculations in the SAR. Typically, this temperature is 125º to 150º F. Spent Fuel Pool temperature is normally maintained well below this point thus allowing time to correct the cooling system malfunction prior to classification.

D-HU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability: Not Applicable

Example Emergency Action Levels:

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

(2) Notification of a security threat determined to be credible per (site-specific security procedure).

(3) Validated notification from the NRC of a threat that involves a potential AIRCRAFT impact on the plant.

Basis:

These events pose a threat to the safety of plant personnel, and possibly to SAFETY-RELATED equipment as well. Security events which do not represent a potential degradation in the level of safety of the plant 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 D-HA1.

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.

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, and the anticipated impact time is greater than 30 minutes or indeterminate. 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. This EAL is met when the threat-related information has been validated in accordance with (site-specific procedure).

Escalation to the Alert emergency classification level would be via IC D-HA1.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

For EAL #1 - 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 [ISFSI]*, Revision 6.

D-HU2

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Natural or destructive phenomena inside the PROTECTED AREA affecting the ability to maintain spent fuel integrity.

Operating Mode Applicability: Not Applicable

Example Emergency Action Levels:

(1) a. **ANY** of the following:

- Seismic event greater than Operating Basis Earthquake (OBE) as indicated by (site-specific indication that a seismic event met or exceeded the OBE limit)
- A tornado strike within the PROTECTED AREA or high winds greater than (site-specific mph)
- EXPLOSION (not due to a HOSTILE ACTION)
- Internal flooding
- Vehicle crash
- FIRE not extinguished within 15 minutes of notification or detection
- Release of a toxic, corrosive, asphyxiant or flammable gas
- (Other site-specific event)

AND

b. The event has the potential to affect, or has affected, equipment necessary to maintain spent fuel integrity.

Basis:

The events described in this IC are of sufficient magnitude to affect or potentially affect systems, structures or components necessary to maintain spent fuel integrity (e.g., spent fuel cooling system).

Escalation of the emergency classification level, if appropriate, will be based on increasing radiation levels via D-AA2 or due to HOSTILE ACTIONS via D-HA1.

First bullet - An Operating Basis Earthquake (OBE) is defined as “an earthquake that could be expected to affect the site of a nuclear reactor, but for which the plant's power production equipment is designed to remain functional without undue risk to public health and safety.” While seismic motion-induced damage may be caused to some portions of the site, but there should be no impact to equipment necessary to maintain spent fuel integrity. Earthquakes of this magnitude will be readily felt by site personnel (e.g., typical lateral accelerations are in excess of 0.08g).

Fourth bullet - This threshold addresses internal flooding events caused by component failures, inadequate configuration control, etc. Flooding, as used in this EAL, describes a condition where water is entering an area faster than installed equipment can remove it, resulting in rising water level within the area. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

Sixth bullet - The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken with the Control Room or other nearby site-specific location to ensure that it is not spurious. An alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. A report from personnel at the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket).

The 15-minute time period starts with a credible notification/report that a FIRE is occurring, or upon verification that a FIRE detection system alarm/actuation is due to a FIRE. Examples of a starting point are:

a. A credible notification/report to the Control Room would be a communications from a member of the plant staff that identifies the observation of a FIRE in a specific location.

   NOTE: In this case, the 15-minute clock to assess the EAL and to extinguish the FIRE starts upon Control Room receipt of the FIRE notification/report.

b. Verification that a FIRE detection system alarm/actuation is due to a FIRE (not a spurious/false alarm) includes either one of the following:

   (1) Control Room (or other nearby site-specific location) receipt of related independent alarm(s) (e.g., FIRE, temperature, deluge system actuation, FIRE pump start, etc.).

   NOTE: In this case, the 15-minute clock to assess the EAL and to extinguish the FIRE starts upon receipt of the independent alarm(s) related to the FIRE.

   (2) On/Near-scene visual confirmation if only a single FIRE/smoke detector has alarmed.

   NOTE: In this case, the 15-minute clock to assess the EAL and to extinguish the FIRE starts upon an on/near-scene confirmation of a FIRE related to the single FIRE/smoke detector that had alarmed.

Seventh bullet - For this threshold to be met, there must be a release of a gas in sufficient quantity that it could reasonably be expected to pose a threat to systems, structures or components necessary to maintain spent fuel integrity (e.g., spent fuel cooling system). This precludes classifications based on small or incidental releases (e.g., handheld fire extinguishers, unexpected releases that are promptly isolated, etc.). It is not intended that a hazards/engineering analysis or air sampling be performed to assess the EAL.

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.

The fact that SCBAs may be worn does not eliminate the need to declare the event.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area.

Eighth bullet - This EAL addresses other site-specific phenomena (such as hurricane, flood or seiche) that have the potential to result loss of spent fuel integrity.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1C

For EAL #1.a – first bullet - This EAL should be based on the capabilities, alarms, and displays of site-specific seismic monitoring equipment. For sites that cannot readily determine when seismic motion has exceeded OBE levels, this EAL statement may be shortened to “Seismic event”.

For EAL #1.a – second bullet - The high wind value should be based on the site-specific UFSAR design basis wind loading for SAFETY-RELATED structures, or the maximum accurate reading available from wind speed instrumentation, whichever is less. There may be several wind speed values available to the Control Room; values may be measured at different elevations and over different time intervals (e.g., instantaneous, 5-minute average, 15-minute average, etc.). The basis document should address which wind speed indications are used for EAL evaluation, and in what priority order.

For EAL #1.a – eighth bullet – Include other site-specific events that meet the Alert criteria presented in Section 3.1.2. Consider significant natural phenomenon that may affect the site (e.g., severe weather events, etc.).

D-HU3

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE.

Operating Mode Applicability: Not Applicable

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-RELATED systems occurs.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a NOUE.

D-AA1

Initiating Condition - ALERT

Any release of gaseous or liquid radioactivity to the environment greater than 200 times the (site-specific effluent release controlling document) limits for 15 minutes or longer.

Operating Mode Applicability: Not Applicable

Example Emergency Action Levels:

Note: In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

(1) Reading on **ANY** effluent radiation monitor greater than (200 times the site-specific effluent controlling document limits) for 15 minutes or longer:

(site-specific monitor list and threshold values corresponding to 200 times the controlling document limits)

(2) Confirmed sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 200 times (site-specific effluent release controlling document limits) for 15 minutes or longer.

Basis:

This IC addresses an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory commitments for an extended period of time (e.g., a significant uncontrolled release).

Defueled 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.

The (site-specific effluent controlling document limit) multiples are specified in D-AU1 and D-AA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the associated release path to the environment has been isolated, the effluent monitor reading is no longer VALID.

Releases should not be prorated or averaged. For example, a release exceeding 600 times release limits for 5 minutes does not meet the EAL.

<u>EAL #1</u>

This EAL addresses radioactivity releases from continuously monitored gaseous or liquid release pathways.

<u>EAL #2</u>

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.).

Developer Notes:

ECL Assignment Attributes: 3.1.2.C

Refer to Appendix A for the detailed basis of the radiological effluent IC/EALs.

Radiological release control limits are typically described in the Offsite Dose Calculation Manual (ODCM), or for plants that have not implemented Generic Letter 89-01, in the Radiological Effluent Technical Specifications (RETS).

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 exceeding 200 times a release control limit. The release control documents typically describe methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values.

Developers should research historical discharge permit setpoints to ensure the applicable monitor's operating range will indicate the highest reasonably expected EAL value (200 times the setpoint). If necessary, establish an EAL value appropriate for conditions when the monitor may be over-ranged by the 200 times value.

Calculations supporting the release rates specified in the EAL values should be provided which quantify expected doses at the Restricted Area Boundary. The major isotope of concern in the permanently defueled condition is Kr-85.

D-AA2

Initiating Condition - ALERT

Rise in radiation levels within the facility that impedes operation of systems required to maintain spent fuel integrity.

Operating Mode Applicability: Not Applicable

Example Emergency Action Levels:

(1) UNPLANNED dose rate greater than 15 mR/hr in **ANY** of the following areas requiring continuous occupancy to maintain control of radioactive material or operation of systems needed to maintain spent fuel integrity:

(site-specific area list)

(2) UNPLANNED Area Radiation Monitor readings or survey results indicate a rise by 100 mR/hr over NORMAL LEVELS that impedes access to **ANY** of the following areas needed to maintain control of radioactive material or operation of systems needed to maintain spent fuel integrity.

(site-specific area list)

Basis:

This IC addresses increased radiation levels that impede necessary access to areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain systems needed to maintain spent fuel integrity. As used here, ‘impede’ includes hindering or interfering, provided that the interference or delay is sufficient to significantly threaten necessary plant access. It is this impaired access that results in the actual or potential substantial degradation of the level of safety of the plant.

This IC does not apply to anticipated temporary increases due to planned events.

Developer Notes:

ECL Assignment Attributes: 3.1.2.C

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

The specified value of 100 mR/hr is arbitrary and may be set to another value for a specific application with appropriate justification.

D-HA1

Initiating Condition - ALERT

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.

Operating Mode Applicability: Not Applicable

Example Emergency Action Levels:

(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA (OCA) as reported by the (site security shift supervision).

(2) A validated notification from NRC of an AIRLINER/LARGE AIRCRAFT attack threat within 30 minutes of the site.

Basis:

This IC addresses the potential for 1) a very rapid progression of events due to a HOSTILE ACTION within the OWNER CONTROLLED AREA (i.e., event could quickly progress to an attack on the PROTECTED AREA), or 2) wide-area damage from an AIRLINER/LARGE AIRCRAFT impact. Either event may require rapid assistance due to the possibility of significant and/or indeterminate damage to equipment, and the potential for casualties. As time and conditions allow, these events require a heightened state of readiness and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering).

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

This IC is 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 within the PROTECTED AREA, 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.

<u>EAL #1</u>

This EAL 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 but within the OWNER CONTROLLED AREA.

<u>EAL #2</u>

This EAL addresses the threat from the impact of an AIRLINER/LARGE 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 AIRLINER/LARGE AIRCRAFT. The status and size of the plane may be provided by NORAD through the NRC.

Only the plant against which the specific threat is made need declare the Alert.

Developer Notes:

ECL Assignment Attributes: 3.1.2.D

D-HA3

Initiating Condition - ALERT

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.

Operating Mode Applicability: Not Applicable

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 EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for an Alert.

9 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: Not* Applicable

Note: Security-related events for ISFSIs are covered under Recognition Category H.

E-HU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: Not applicable

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 is intended to describe an event that results in damage to a loaded cask CONFINEMENT BOUNDARY. The concerns are 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 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 extrapolated from a dose rate measured at some distance from the cask.

Developer Notes:

ECL Assignment Attributes: 3.1.1.B, 3.1.1.C and 3.1.1.D

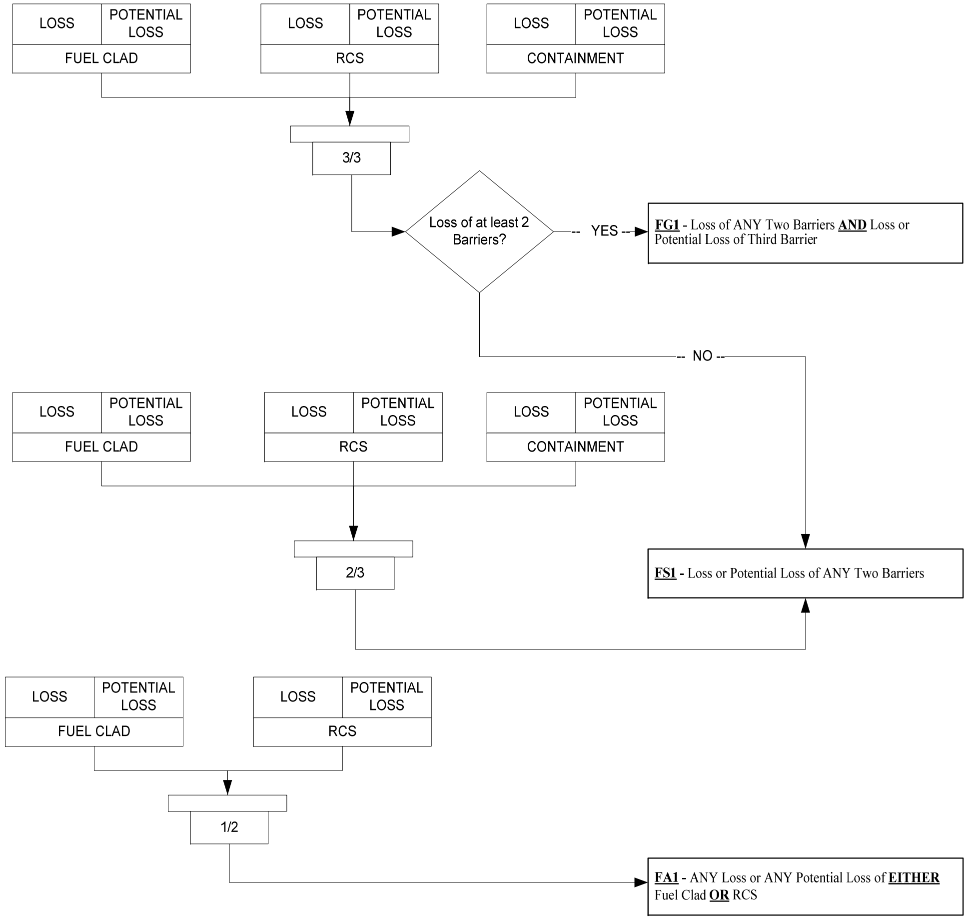
The results of the ISFSI Safety Analysis Report (SAR) per NUREG 1536 or SAR referenced in the cask(s) Certificate of Compliance and the related NRC Safety Evaluation Report identify natural phenomena events and accident conditions that could potentially affect the CONFINEMENT BOUNDARY. This EAL addresses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

10 FISSION PRODUCT BARRIER ICS/EALS

Table 10-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, Hot Standby, Startup, Hot Shutdown*
SITE AREA EMERGENCY	
FS1	Loss or Potential Loss of any two barriers. *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*
GENERAL EMERGENCY	
FG1	Loss of any two barriers and Loss or Potential Loss of the third barrier. *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*



See Table 11-F-2 for BWR EALs
See Table 11-F-3 for PWR EALs

Note: The logic flow diagram is for use by developers and is not required for site-specific implementation.

NOTES

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. NOUE ICs associated with the RCS and Fuel Clad Barriers are addressed under the System Malfunction ICs.

- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria unless there is an event in progress requiring mitigation by the Containment Barrier.

Table 10-F-2: BWR 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. Reactor Coolant Activity Level		**1. Primary Containment Pressure**		**1. Primary Containment Conditions**	
A. Reactor coolant activity greater than (site-specific value)	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) and rising **OR** B. (site-specific explosive mixture) exists inside primary containment **OR** C. RPV pressure and suppression pool temperature cannot be maintained below the HCTL

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. RPV Water Level		**2. RPV Water Level**		**2. RPV Water Level**	
A. Primary containment flooding required	A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) following depressurization of the RPV or cannot be determined	A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) following depressurization of the RPV or cannot be determined	Not Applicable	Not Applicable	A. Primary containment flooding required
3. Not Applicable		**3. RCS Leak Rate**		**3. Primary Containment Isolation Failure**	

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
Not Applicable	Not Applicable	A. Indication of an UNISOLABLE break in **ANY** of the following: (site-specific systems with potential for high-energy line breaks) **OR** B. Emergency RPV Depressurization is required	A. UNISOLABLE primary system leakage outside primary containment that results in exceeding **EITHER** of the following: 1. Max Normal Operating Temperature **OR** 2. Max Normal Operating Area Radiation Level	A. Failure of valves in **ANY** one line to close **AND** 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 outside primary containment that results in exceeding **EITHER** of the following: 1. Max Safe Operating Temperature. **OR** 2. Max Safe Operating Area Radiation Level	Not Applicable

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
4. Primary Containment Radiation Monitoring		**4. Primary Containment Radiation Monitoring**		**4. Primary Containment Radiation Monitoring**	
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	Not Applicable	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier	Not Applicable	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier	Not Applicable

Basis Information For
BWR EAL Fission Product Barrier Table 10-F-2

FUEL CLAD BARRIER THRESHOLDS:

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

1. Reactor Coolant Activity Level

Loss 1.A

The site-specific value corresponds to 300 μCi/gm I-131 equivalent. Assessment by the NEI 99-01 EAL Task Force indicates that this amount of reactor coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no Potential Loss threshold associated with this item.

Developer Notes:

The threshold value should be expressed as either a dose rate measured on the sample or radioactivity concentration such as μCi/gm or μCi/cc.

2. RPV Water Level

Loss 2.A

The Loss threshold represents the EOP requirement for primary containment flooding. This is identified in the BWROG EPGs/SAGs when the phrase, “Primary Containment Flooding Is Required," appears. Since a site-specific RPV water level is not specified here, the Loss threshold phrase, “Primary Containment Flooding Required,” also accommodates the EOP need to flood the primary containment when RPV water level cannot be determined and core damage due to inadequate core cooling is believed to be occurring.

Potential Loss 2.A

The RPV water level threshold is the same as RCS barrier Loss threshold 2.A and corresponds to the site-specific water level at the top of the active fuel and to a challenge to core cooling. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is 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). 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 the RPV has been depressurized and the operator has been able to assess the capability of low-pressure injection sources to restore RPV water level.

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 SA2, SS2 and SG2 will dictate the need for emergency classification.

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, “Primary Containment Flooding Is Required,” should be modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs (e.g., drywell flooding required, etc.).

<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 Monitoring

Loss 4.A

The site-specific reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell.

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage.

This value is higher than that specified for RCS barrier Loss threshold 4.A. Thus, this threshold indicates a loss of both Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with primary containment radiation indications.

Developer Notes:

The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 or the calculated concentration equivalent to the clad damage used in threshold 1 into the drywell atmosphere.

Caution: It is important to recognize that in the event the radiation monitor is sensitive to shine from the reactor vessel or piping, spurious readings will be present and another indicator of fuel clad damage is necessary or compensated for in the threshold value.

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.

Developer Notes:

To ensure consistent classifications, any specified thresholds should be approximately equivalent, in relative threat, to the thresholds provided in the same column. Use the basis information from equivalent or similar thresholds to determine the relative threat.

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. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

There is no Potential Loss threshold associated with Emergency Director judgment.

RCS BARRIER THRESHOLDS:

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

1. Primary Containment Conditions

Loss 1.A

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

There is no Potential Loss threshold associated with this primary containment condition.

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 site-specific RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold 2.A. Thus, this threshold indicates a Loss of RCS barrier and Potential Loss of Fuel Clad barrier and 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, that 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). There is no Potential Loss threshold associated with this RPV water level.

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 SA2, SS2 and SG2 will dictate the need for emergency classification.

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

3. RCS Leak Rate

Loss Threshold 3.A

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.

Loss Threshold 3.B

Plant symptoms requiring Emergency RPV Depressurization per the EOPs are indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is required, 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.

Potential Loss Threshold 3.A

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, RCIC, HPCI, 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.

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:

Loss Threshold 3.A

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 system.

Large high-energy line breaks such as Main Steam Line (MSL), High Pressure Coolant Injection (HPCI), Feedwater, Reactor Water Cleanup (RWCU), Isolation Condenser (IC) or Reactor Core Isolation Cooling (RCIC) that are UNISOLABLE represent a significant loss of the RCS barrier.

4. Primary Containment Radiation Monitoring

Loss 4.A

(site-specific reading) is a value which indicates the release of reactor coolant to the primary containment.

This reading will be less than that specified for Fuel Clad Barrier Loss threshold 4.A. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that value specified by Fuel Clad Barrier threshold, fuel damage would also be indicated.

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

Developer Notes:

The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., maximum allowed by T/S) into the drywell atmosphere.

However, if the site-specific physical location of the primary containment radiation monitor is such that radiation from a cloud of released RCS gases could not be distinguished from radiation from adjacent piping and components containing elevated reactor coolant activity, this threshold should be omitted and other site-specific indications of RCS leakage substituted, if available.

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.

Developer Notes:

To ensure consistent classifications, any specified thresholds should be approximately equivalent, in relative threat, to the thresholds provided in the same column. Use the basis information from equivalent or similar thresholds to determine the relative threat.

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 RCS barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

There is no Potential Loss threshold associated with Emergency Director judgment.

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 only as discriminators for escalation from an Alert to a Site Area Emergency or a General Emergency.

1. Primary Containment Conditions

<u>Loss 1.A and 1.B</u>

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 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.

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

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.

<u>Potential Loss 1.B</u>

If hydrogen concentration reaches or exceeds the lower flammability limit 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.

<u>Potential Loss 1.C</u>

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

- 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 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. For Mk I/II containments the deflagration limits are "6% hydrogen and 5% oxygen in the drywell or suppression chamber". For Mk III containments, the limit is the "Hydrogen Deflagration Overpressure Limit". The threshold term "explosive mixture" is synonymous with the EPG/SAG "deflagration limits".

Potential Loss 1.C

Since the HCTL is defined assuming a range of suppression pool water levels as low as the elevation of the downcomer openings in Mk I/II containments or 2 ft above the elevation of the horizontal vents in a Mk III containment, it is unnecessary to consider separate Containment barrier Loss or Potential Loss thresholds for abnormal suppression pool water level conditions.

2. RPV Water Level

There is no Containment Loss threshold associated with RPV water level.

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold.1.A. The Potential Loss requirement for Primary Containment Flooding indicates adequate core cooling cannot be restored and maintained and that core damage is possible. BWR EPGs/SAGs specify the conditions that require primary containment flooding. When primary containment flooding is required, the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to restore and maintain adequate core cooling.

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:

Severe Accident Guidelines (SAGs) direct the operators to perform Primary Containment Flooding when RPV water level cannot be restored and maintained greater than the Minimum Steam Cooling RPV Water Level or RPV water level cannot be determined with indication that core damage is occurring.

3. Primary Containment Isolation Failure

These thresholds address incomplete containment isolation that allows 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. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the RPV, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

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 secondary containment and/or 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.

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 or bypass.

4. Primary Containment Radiation Monitoring

There is no Loss threshold associated with primary containment radiation monitoring.

Potential Loss 4.A

The value indicates significant fuel damage well in excess of that required for loss of RCS and Fuel Clad.

Regardless of whether primary containment is challenged, this amount of activity in the primary containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted.

Developer Notes:

A major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates that such conditions do not exist when the amount of clad damage is less than 20%. Unless there is a (site-specific) analysis justifying a higher value, it is recommended that a radiation monitor reading corresponding to 20% fuel clad damage be specified here.

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.

Developer Notes:

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.

To ensure consistent classifications, any specified thresholds should be approximately equivalent, in relative threat, to the thresholds provided in the same column. Use the basis information from equivalent or similar thresholds to determine the relative threat.

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 Containment barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

There is no Potential Loss threshold associated with Emergency Director judgment.

Table 10-F-3: PWR 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. RCS/reactor vessel level less than (site-specific level).	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage.	A. Operation of a standby charging (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. Leaking 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 as indicated by (site-specific indications).	Not Applicable	A. Inadequate RCS heat removal capability via steam generators 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 I-131 dose equivalent).	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).

<table>
<tr><th colspan="2"><u>Fuel Clad Barrier</u></th><th colspan="2"><u>RCS Barrier</u></th><th colspan="2"><u>Containment Barrier</u></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 colspan="2">4. Containment Integrity or Bypass</td><td colspan="2">4. Containment Integrity or Bypass</td><td colspan="2">4. Containment Integrity or Bypass</td></tr>
<tr><td>Not Applicable</td><td>Not Applicable</td><td>Not Applicable</td><td>Not Applicable</td><td>A. Containment isolation is required
AND
EITHER of the following:
1. UNPLANNED rise in radiation monitor readings outside of containment that indicate a loss of containment integrity
OR
2. UNISOLABLE pathway from the containment to the environment exists
OR
B. Indications of RCS leakage outside of containment.</td><td>A. Containment pressure greater than (site-specific value) and rising
OR
B. Explosive mixture exists inside containment
OR
C. 1. Pressure greater than (site-specific containment depressurization actuation setpoint)
AND
2. Less than one full train of (site-specific containment depressurization equipment operating per design).</td></tr>
<tr><td colspan="2">5. Other Indications</td><td colspan="2">5. Other Indications</td><td colspan="2">5. Other Indications</td></tr>
<tr><td>A. (site-specific as applicable).</td><td>A. (site-specific as applicable).</td><td>A. (site-specific as applicable).</td><td>A. (site-specific as applicable).</td><td>A. (site-specific as applicable).</td><td>A. (site-specific as applicable).</td></tr>
</table>

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
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.	Not Applicable	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	Not Applicable	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	Not Applicable

Basis Information For
PWR EAL Fission Product Barrier Table 10-F-3

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.

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

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

Developer Notes:

Enter the site-specific reactor vessel water level value(s) used by EOPs to identify a degraded core cooling condition. The reactor vessel level that corresponds to the top of active fuel may also be used.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines:

The value in Potential Loss 1.A is the reactor vessel level(s) used for the Core Cooling Orange Path. Developers may also include a threshold the same as, or similar to, "Core Cooling Orange Entry Conditions Met".

2. Inadequate Heat Removal

<u>Loss 2.A</u>

This condition indicates an extreme challenge to core cooling, i.e., temperature sufficient to cause significant superheating of reactor coolant within the core.

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

This condition indicates a severe challenge to core cooling, i.e., temperature sufficient to allow the onset of heat-induced cladding damage.

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

This condition indicates an extreme challenge to the secondary heat sink (i.e., ability to remove RCS heat using the steam generators) due to inadequate steam generator feed water flow and/or water inventory. This condition represents a potential loss of the Fuel Clad Barrier. The heat sink must be required for this threshold to be considered VALID.

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 the loss of secondary heat sink may result in RCS heatup sufficient to damage fuel cladding and increase RCS pressure to the point where mass will be lost from the system.

Developer Notes:

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.

Potential Loss 2.A – Enter a site-specific temperature value that corresponds to core conditions at the onset of heat-induced cladding damage. 700^{o}F may also be used.

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.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines:

- The value in Loss 2.A is the CET value used in the Core Cooling Red Path. Developers may also include a threshold the same as, or similar to, “Core Cooling Red Entry Conditions Met”.
- The value in Potential Loss 2.A is the CET value used in the Core Cooling Orange Path. Developers may also include a threshold the same as, or similar to, “Core Cooling Orange Entry Conditions Met”.

Potential Loss 2.B – An extreme challenge to RCS heat removal means that heat removal via the steam generators has (or soon will) become ineffective. An extreme challenge exists if the minimum level in the minimum number of steam generators cannot be maintained. Emergency (auxiliary) feedwater flow and/or steam generator level values should be determined based on the above description of the condition.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines:

The values in Potential Loss 2.B are those used for the Heat Sink Red Path. Developers may also include a threshold the same as, or similar to, “Heat Sink Red Entry Conditions Met”.

3. RCS Activity / Containment Radiation

Loss 3.A

The radiation monitor reading corresponds to an instantaneous release of the radioactive material inventory of the reactor coolant system (i.e., all the RCS coolant mass) into the containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. This radioactivity concentration is several times larger than that allowed by Technical Specifications and is the same concentration that defines a loss of the Fuel Clad Barrier in threshold 3.B.

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.

Loss 3.B

This value indicates that RCS activity is 300 μCi/gm dose equivalent I-131. This amount of reactor coolant activity is well above 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 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:

Potential Loss 3.A - The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS activity equal to 300 μCi/gm dose equivalent I-131, into the containment atmosphere.

Potential Loss 3.B - The reading should be calculated assuming RCS activity equals 300 μCi/gm dose equivalent I-131. For plants that have the ability to detect this level of fuel clad damage with installed radiation monitors, consideration should be given to using radiation monitor readings in addition to sample analysis results for this threshold.

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

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 a loss or potential loss of the Fuel Clad Barrier. Consider use of area, process and airborne radiation monitor readings.

Developer Notes:

To ensure consistent classifications, any specified thresholds should be approximately equivalent, in relative threat, to the thresholds provided in the same column. Use the basis information from equivalent or similar thresholds to determine the relative threat.

6. Emergency Director Judgment

<u>Loss 6.A</u>

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

There is no Potential Loss threshold associated with Emergency Director Judgment.

RCS BARRIER THRESHOLDS:

The Reactor Coolant System (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 RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). The RCS leak must be UNISOLABLE. 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. 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.

Potential Loss 1.A

This threshold is based on an RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The RCS leak must be UNISOLABLE. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. 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.

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:

Potential Loss 1.A - For plants with low capacity charging pumps (i.e., <50 gpm), a 50 gpm RCS leak rate may be used as an alternate Potential Loss threshold value.

Actuation of the ECCS may also be referred to as Safety Injection actuation or other appropriate site-specific term.

Potential Loss 1.B - For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines:

The values in Potential Loss 1.B are those used for the RCS Integrity Red Path. Developers may also include a threshold the same as, or similar to, “RCS Integrity Red Entry Conditions Met”.

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 secondary heat sink (i.e., ability to remove RCS heat using the steam generators) due to inadequate steam generator feed water flow and/or water inventory. This condition represents a potential loss of the RCS Barrier. The heat sink must be required for this threshold to be considered VALID.

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 the loss of secondary heat sink may result in RCS heatup sufficient to damage fuel cladding and increase RCS pressure to the point where mass will be lost from the system.

Developer Notes:

An extreme challenge to RCS heat removal means that heat removal via the steam generators has (or soon will) become ineffective. An extreme challenge exists if the minimum level in the minimum number of steam generators cannot be maintained. Emergency (auxiliary) feedwater flow and/or steam generator level values should be determined based on the above description of the condition.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines:

The values in Potential Loss 2.A are those used for the Heat Sink Red Path. Developers may also include a threshold the same as, or similar to, “Heat Sink Red Entry Conditions Met”.

3. RCS Activity / Containment Radiation

Loss 3.A

The radiation monitor reading corresponds to an instantaneous release of the radioactive material inventory of the reactor coolant system (i.e., all the RCS coolant mass) into the containment, assuming that RCS activity is at 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.

Developer Notes:

The reading should be calculated 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 containment atmosphere. Using RCS activity at Technical Specification allowable limits aligns this threshold with IC SU4. Also, RCS activity at his 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.

If the site-specific physical location of the containment radiation monitor(s) is such that radiation from a cloud of released RCS gases could not be distinguished from radiation emanating from piping and components containing elevated reactor coolant activity, this threshold should be omitted and other site-specific indications of RCS leakage substituted.

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

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.

Developer Notes:

To ensure consistent classifications, any specified thresholds should be approximately equivalent, in relative threat, to the thresholds provided in the same column. Use the basis information from equivalent or similar thresholds to determine the relative threat.

6. Emergency Director Judgment

<u>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 lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

There is no Potential Loss threshold associated with Emergency Director judgment.

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.

1. RCS or SG Tube Leakage

Loss 1.A

This threshold addresses a leaking Steam Generator (SG) that is also FAULTED outside of containment. This condition represents a bypass of the containment barrier. FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP.

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve do not meet the intent of this threshold. Such releases may occur 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.

If the main condenser is available, there may be minor releases via air ejectors, gland seal exhausters, and other similar pathways. These types of releases do not meet the intent of this threshold; rather, they are assessed using the Category A ICs dealing with radiological releases.

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

P-to-S Leak Rate	Affected SG is FAULTED Outside of Containment? Yes	No
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU5	Unusual Event per SU5
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1	Alert per FA1
Requires an automatic or manual ECCS (SI) 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:

A steam generator power operated relief valve may also be referred to as an atmospheric steam dump valve or other appropriate site-specific term.

Developers may include an additional site-specific threshold(s) to address prolonged steam releases necessitated by operational considerations if EOPs could require that a leaking steam generator be used to support plant cooldown.

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.

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.

Developer Notes:

Potential Loss 2.A.1 – List site-specific criteria for entry into core cooling restoration procedure. A 1,200^{o}F reading on the CETs may also be used.

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.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines:

The values in Potential Loss 2.A.1 are those used in the Core Cooling Red Path. Developers may also include a threshold the same as, or similar to, "Core Cooling Red Entry Conditions met for 15 minutes or longer".

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

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 the radioactive material inventory of the reactor coolant system (i.e., all the RCS coolant mass) into the containment, assuming that 20% of the fuel cladding has failed. This level of assumed fuel damage is well beyond that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

Approximately 20% of the fuel cladding must fail in order for there to be a major release of radioactivity requiring offsite protection actions. For this condition to occur, 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:

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, it is recommended that the calculated radiation monitor reading correspond to 20% fuel clad failure.

4. Containment Integrity or Bypass

Loss 4.A

This threshold addresses a situation where containment isolation is required and one of two conditions exists.

4.A.1 – Despite the containment isolation, radioactive material in the containment is escaping to an in-plant location outside of containment. For example, radioactive material may be entering an auxiliary building due to containment leakage (from a penetration) or through leakage in an in-service system (from a mechanical connection). Leakage of this type will be most readily detected by in-plant radiation monitors. Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the middle piping run of Figure 10-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.

4.A.2 – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment to a point outside of the containment where the material can enter, or become entrained in, a ventilation system flow path that ultimately exhausts to the environment. Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 10-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.

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

Refer to the bottom piping run of Figure 10-F-4. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then threshold 4.B would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors and cause threshold 4.A.1 to be met as well.

This threshold is not applicable to conditions involving primary-to-secondary (i.e., steam generator) leakage. The status of the containment barrier under those conditions is assessed using Loss Threshold 1.A.

Loss 4.B

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 10-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 and cause threshold 4.A.1 to be met as well.

To ensure proper escalation of the emergency classification, the RCS mass being lost 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.

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 containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

Developer Notes:

For Loss 4.A.1 – Developers may include a list of site-specific radiation monitors to assist with better defining this threshold. For example, the threshold might read "UNPLANNED rise in one or more of the following radiation monitors outside containment indicating a loss of containment integrity (site-specific list of monitors)".

For 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. Developers may also include a threshold the same as, or similar to, "Containment Red Entry Conditions Met".

For 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 is not applicable to the U.S. Evolutionary Power Reactor (EPR) design.

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.

Developer Notes:

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.

To ensure consistent classifications, any specified thresholds should be approximately equivalent, in relative threat, to the thresholds provided in the same column. Use the basis information from equivalent or similar thresholds to determine the relative threat.

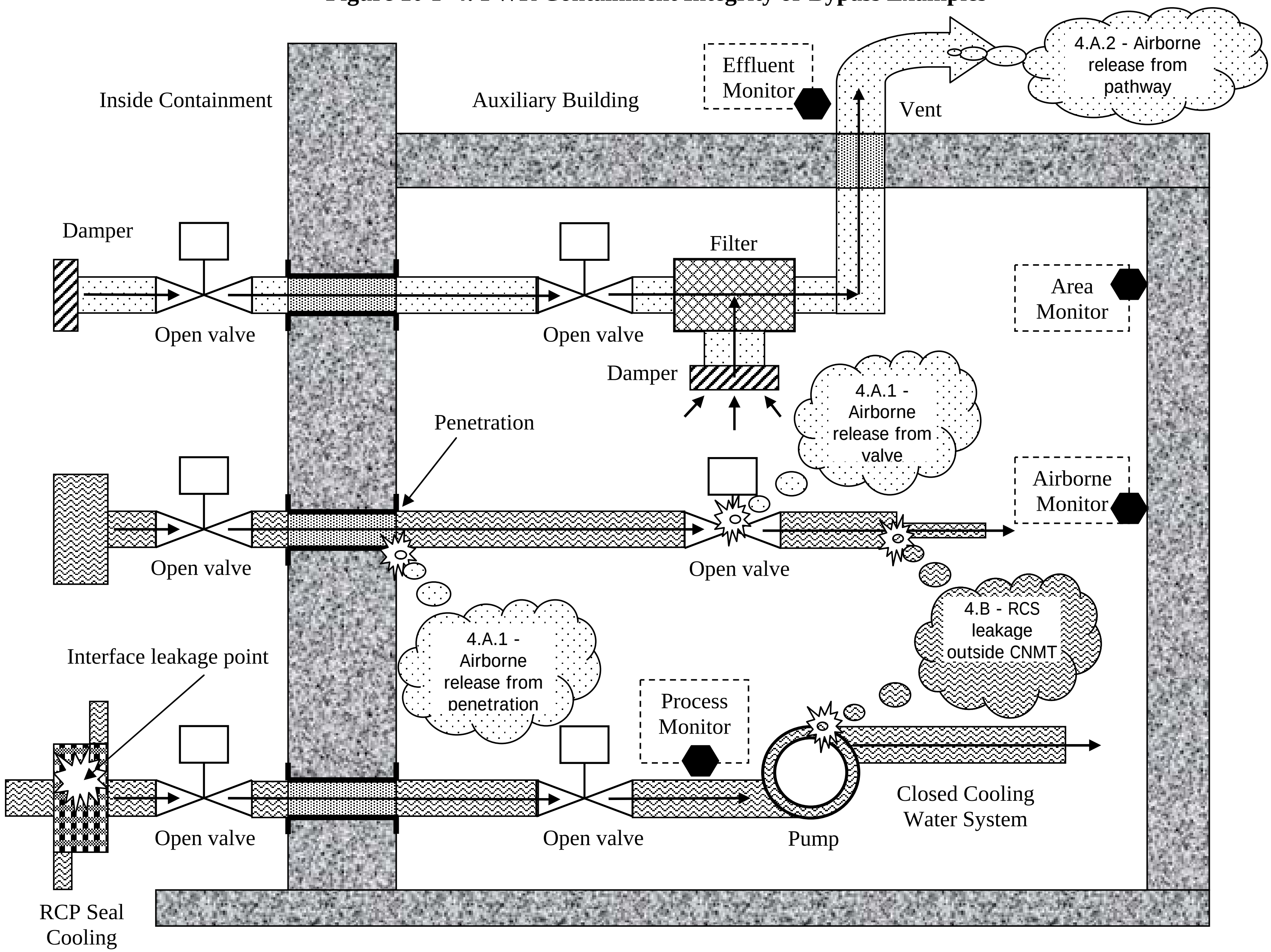
6. Emergency Director Judgment

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. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

There is no Potential Loss threshold associated with Emergency Director judgment.

Figure 10-F-4: PWR Containment Integrity or Bypass Examples



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11 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 which indicates a potential degradation in the level of safety of the plant. *Op. Modes: All*	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat. *Op. Modes: All*	**HS1** HOSTILE ACTION within the PROTECTED AREA. *Op. Modes: All*	**HG1** HOSTILE ACTION resulting in loss of key safety functions or damage to spent fuel. *Op. Modes: All*
HU2 Natural or destructive events challenging design limits within the PROTECTED AREA. *Op.* Modes*: All*	**HA2** Natural or destructive events affecting a SAFETY-RELATED structure or area, or resulting in degraded SAFETY-RELATED system performance. *Op.* Modes*: All*		
HU3 FIRE in SAFETY-RELATED structures or areas not extinguished within 15 minutes. *Op.* Modes*: All*	**HA3** FIRE resulting in VISIBLE DAMAGE to a SAFETY-RELATED structure or area, or resulting in degraded SAFETY-RELATED system performance. *Op.* Modes*: All*		
HU4 Release of a toxic, corrosive, asphyxiant, or flammable gas AFFECTING NORMAL PLANT OPERATIONS. *Op.* Modes*: All*	**HA4** Release of a toxic, corrosive, asphyxiant or flammable gas resulting in degraded SAFETY-RELATED system performance. *Op. Modes: All*		

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	HA5 Control Room has been evacuated. *Op.* Modes*: All*	**HS5** Control Room has been evacuated and control of key safety functions has not been established. *Op. Modes: All*	
HU6 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE. *Op. Modes: All*	**HA6** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert. *Op. Modes: All*	**HS6** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency. *Op.* Modes*: All*	**HG6** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency. *Op.* Modes*: All*

HU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability: All

Example Emergency Action Levels:

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

(2) Notification of a security threat determined to be credible per (site-specific procedure).

(3) Validated notification from the NRC of a threat that involves a potential aircraft impact on the plant.

Basis:

These events pose a threat to the safety of plant personnel, and possibly to SAFETY-RELATED equipment as well. Security events which do not represent a potential degradation in the level of safety of the plant 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.

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, and the anticipated impact time is greater than 30 minutes or indeterminate. 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. This EAL is met when the threat-related information has been validated in accordance with (site-specific procedure).

Escalation to the Alert emergency classification level would be via IC HA1.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

For EAL #1 - 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 [ISFSI]*, Revision 6.

HU2

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Natural or destructive events challenging design limits within the PROTECTED AREA.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by (site-specific indication that a seismic event met or exceeded the OBE limit).

(2) A tornado strike within the PROTECTED AREA or high winds greater than (site-specific mph).

(3) (Other site-specific natural or destructive events that may challenge design limits within the PROTECTED AREA).

Basis:

These events represent a potential degradation of the level of safety of the plant. The distinction between IC HU2 and IC HA2, and their associated EALs, is that the magnitude of an HA2 event is severe enough to have caused either VISIBLE DAMAGE to a SAFETY-RELATED structure or area, or damage sufficient to degrade the performance of more than one SAFETY-RELATED train or more than one SAFETY-RELATED system. Degraded performance would be indicated if the affected trains and/or systems are unable to perform their intended function.

Escalation of the emergency classification level, if appropriate, would be based on IC HA2.

<u>EAL #1</u>

An Operating Basis Earthquake (OBE) is defined as “an earthquake that could be expected to affect the site of a nuclear reactor, but for which the plant's power production equipment is designed to remain functional without undue risk to public health and safety.” While seismic motion-induced damage may be caused to some portions of the site, but there should be no impact on the ability of SAFETY-RELATED equipment to operate. Earthquakes of this magnitude will be readily felt by site personnel (e.g., typical lateral accelerations are in excess of 0.08g).

<u>EAL #2</u>

A tornado striking (touching down) within the Protected Area warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

EAL #3

(Basis for inclusion of other site-specific events).

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.C

For EAL #1 - This EAL should be based on the capabilities, alarms and displays of site-specific seismic monitoring equipment.

For sites that cannot readily determine when seismic motion has exceeded OBE levels, the following alternate EAL wording may be used:

> Seismic event AFFECTING NORMAL PLANT OPERATIONS.

This wording may also be used during periods when a seismic monitoring system is out-of-service for maintenance or repair.

For EAL #2 - The high wind value should be based on the site-specific UFSAR design basis wind loading for SAFETY-RELATED structures, or the maximum accurate reading available from wind speed instrumentation, whichever is less. There may be several wind speed values available to the Control Room; values may be measured at different elevations and over different time intervals (e.g., instantaneous, 5-minute average, 15-minute average, etc.). The basis section should address which wind speed indications are used for EAL evaluation, and in what priority order.

For EAL #3 – Include other site-specific events that meet the Unusual Event criteria presented in Section 3.1.1.

HU3

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

FIRE in SAFETY-RELATED structures or areas not extinguished within 15 minutes.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) a. FIRE within **ANY** of the following:

(site-specific list of SAFETY-RELATED structures and areas)

AND

b. FIRE is not extinguished within 15 minutes of **EITHER** of the following:

- Control Room notification of a FIRE
- Verified FIRE detection system alarm/actuation

Basis:

This EAL addresses a FIRE of sufficient size and duration to be a precursor to damage to a SAFETY-RELATED system, structure or component. The intent of the 15-minute duration criterion is rule out a small FIRE that can be readily extinguished (e.g., in a waste paper basket).

Structures and areas requiring an EAL assessment are limited to those housing SAFETY-RELATED equipment or otherwise needed for performance of a SAFETY-RELATED function, and immediately adjacent areas. This excludes FIRES of no safety significance.

The 15-minute time period starts with a credible notification/report that a FIRE is occurring, or upon verification that a FIRE detection system alarm/actuation is due to a FIRE. Examples of a starting point are:

a. A credible notification/report to the Control Room would be a communication from a member of the plant staff that identifies the observation of a FIRE in a specific location.

NOTE: In this case, the 15-minute clock to assess the EAL and to extinguish the FIRE starts upon Control Room receipt of the FIRE notification/report.

b. Verification that a FIRE detection system alarm/actuation is due to a FIRE (not a spurious/false alarm) includes either one of the following:

(1) Control Room (or other nearby site-specific location) receipt of related independent alarm(s) (e.g., FIRE, temperature, deluge system actuation, FIRE pump start, etc.).

NOTE: In this case, the 15-minute clock to assess the EAL and to extinguish the FIRE starts upon receipt of the independent alarm(s) related to the FIRE.

(2) On/Near-scene visual confirmation if only a single FIRE/smoke detector has alarmed.

NOTE: In this case, the 15-minute clock to assess the EAL and to extinguish the FIRE starts upon an on/near-scene confirmation of a FIRE related to the single FIRE/smoke detector that had alarmed.

Escalation of this emergency classification level, if appropriate, would be based on IC HA3.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

The site-specific list of SAFETY-RELATED structures and areas should specify those structures or areas that contain SAFETY-RELATED systems, components or functions. Additionally, those structures immediately adjacent to SAFETY-RELATED structures and areas should be included due to the potential for the fire to spread to a SAFETY-RELATED structure or area.

HU4

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Release of a toxic, corrosive, asphyxiant, or flammable gas AFFECTING NORMAL PLANT OPERATIONS.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Release of a toxic, corrosive, asphyxiant or flammable gas in an amount AFFECTING NORMAL PLANT OPERATIONS.

(2) Personnel inside the PROTECTED AREA are directed to evacuate or take shelter due to offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

Basis:

For EAL #1 to be met, there must be a release of a gas in sufficient quantity to cause events that lead to a change to the current reactor power level or entry into an emergency operating procedure. This precludes classifications based on small or incidental releases (e.g. handheld fire extinguishers) and releases that are promptly isolated. It is not intended that a hazards/engineering analysis or air sampling be performed to assess the EAL.

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. This EAL does not apply to intentional inerting of containment (BWR only).

EAL #2 addresses a hazardous materials event originating at an offsite location. If site/shift management order an evacuation or sheltering of personnel within the PROTECTED AREA, then a NOUE declaration is warranted.

Escalation of this emergency classification level, if appropriate, would be based on IC HA4.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

HU6

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE.

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-RELATED systems occurs.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a NOUE.

HA1

Initiating Condition - ALERT

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.

Operating Mode Applicability: All

Example Emergency Action Levels:

(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 AIRLINER/LARGE AIRCRAFT attack threat within 30 minutes of the site.

Basis:

This IC addresses the potential for 1) a very rapid progression of events due to a HOSTILE ACTION within the OWNER CONTROLLED AREA (i.e., the event could quickly progress to an attack on the PROTECTED AREA), or 2) wide-area damage from an AIRLINER/LARGE AIRCRAFT impact. Either event will require rapid assistance due to the possibility for significant and/or indeterminate damage to equipment, and casualties among the plant staff.

As time and conditions allow, these events require a heightened state of readiness and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering).

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

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 within the PROTECTED AREA, 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 but within the OWNER CONTROLLED AREA.

EAL #2 addresses the threat from the impact of an AIRLINER/LARGE 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 AIRLINER/LARGE AIRCRAFT. The status and size of the plane may be provided by NORAD through the NRC.

Developer Notes:

ECL Assignment Attributes: 3.1.2.D

HA2

Initiating Condition - ALERT

Natural or destructive events affecting a SAFETY-RELATED structure or area, or resulting in degraded SAFETY-RELATED system performance.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) a. **ANY** of the following:

- Seismic event greater than Operating Basis Earthquake (OBE) as indicated by (site-specific indication that a seismic event met or exceeded the OBE limit)
- A tornado strike within the PROTECTED AREA or high winds greater than (site-specific mph)
- EXPLOSION (not due to a HOSTILE ACTION)
- Internal flooding
- Turbine failure-generated PROJECTILES
- Vehicle crash
- (Other site-specific event)

AND

b. **ANY** of the following:

- VISIBLE DAMAGE to ANY of the following (site-specific list of SAFETY-RELATED structures and areas)
- Control Room indication of degraded performance of more than one train of a SAFETY-RELATED system or more than one SAFETY-RELATED system
- Damage report of sufficient magnitude to conclude that more than one train of a SAFETY-RELATED system or more than one SAFETY-RELATED system cannot perform their intended design function

Basis:

These events represent an actual or potential substantial degradation of the level of safety of the plant. The distinction between IC HU2 and IC HA2, and their associated EALs, is that the magnitude of an HA2 event is severe enough to have caused either VISIBLE DAMAGE to a SAFETY-RELATED structure or area, or damage sufficient to degrade the performance of more than one SAFETY-RELATED train or more than one SAFETY-RELATED system. Degraded performance would be indicated if the affected trains and/or systems are unable to perform their intended function.

VISIBLE DAMAGE is used to differentiate between an event that causes minor damage and one that has the potential to damage SAFETY-RELATED equipment. The declaration of an Alert and the activation of the Technical Support Center and Operational Support Center will provide the Emergency Director with the resources necessary to perform detailed damage assessments.

Escalation of this emergency classification level would be based on the System Malfunction ICs of Fission Product Barrier Matrix.

EAL #1.a – first bullet

An Operating Basis Earthquake (OBE) is defined as "an earthquake that could be expected to affect the site of a nuclear reactor, but for which the plant's power production equipment is designed to remain functional without undue risk to public health and safety." While seismic motion-induced damage may be caused to some portions of the site, but there should be no impact on the ability of SAFETY-RELATED equipment to operate. Earthquakes of this magnitude will be readily felt by site personnel (e.g., typical lateral accelerations are in excess of 0.08g).

EAL #1.b – second bullet

A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

EAL #1.a – third bullet

A release of steam from a steamline (or a pressurized high temperature water line) is not, in and of itself, evidence of an explosion; this determination should based on the failure mode and the resulting potential for damage to surrounding systems, structures and components.

The Emergency Director also needs to consider any security aspects of the EXPLOSION.

EAL #1.a – fourth bullet

This EAL addresses internal flooding events caused by component failures, inadequate configuration control, outage activities, etc. Flooding, as used in this EAL, describes a condition where water is entering an area faster than installed equipment can remove it, resulting in rising water level within the area. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

EAL #1.a – seventh bullet

(Basis for inclusion of other site-specific events).

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

For EAL #1.a – first bullet - This EAL should be based on the capabilities, alarms, and displays of site-specific seismic monitoring equipment.

For sites that cannot readily determine when seismic motion has exceeded OBE levels, this EAL statement may be shortened to "Seismic event". This wording may also be used during periods when a seismic monitoring system is out-of-service for maintenance or repair.

For EAL #1.a – second bullet - The high wind value should be based on the site-specific UFSAR design basis wind loading for SAFETY-RELATED structures, or the maximum accurate reading available from wind speed instrumentation, whichever is less. There may be several wind speed values available to the Control Room; values may be measured at different elevations and over different time intervals (e.g., instantaneous, 5-minute average, 15-minute average, etc.). The basis document should address which wind speed indications are used for EAL evaluation, and in what priority order.

For EAL #1.a – seventh bullet – Include other site-specific events that meet the Alert criteria presented in Section 3.1.2. Consider significant natural phenomenon that may affect the site (e.g., severe weather events, etc.).

For EAL #1.b - The site-specific list of SAFETY-RELATED structures and areas should specify those structures or areas that contain SAFETY-RELATED systems, components or functions required for safe shutdown of the plant. The site-specific Safe Shutdown Analysis should be consulted for this list of areas.

HA3

Initiating Condition - ALERT

FIRE resulting in VISIBLE DAMAGE to a SAFETY-RELATED structure or area, or resulting in degraded SAFETY-RELATED system performance.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) FIRE resulting in **ANY** of the following:

- VISIBLE DAMAGE to **ANY** of the following:

  (site-specific list of SAFETY-RELATED structures and areas).

- Control Room indication of degraded performance of more than one train of a SAFETY-RELATED system or more than one SAFETY-RELATED system.

- Damage report of sufficient magnitude to conclude that more than one train of a SAFETY-RELATED system or more than one SAFETY-RELATED system cannot perform their intended design function.

Basis:

This IC represents an actual or potential substantial degradation of the level of safety of the plant. The distinction between IC HU3 and IC HA3 is that the magnitude of an HA3 event is severe enough to have caused either VISIBLE DAMAGE to a SAFETY-RELATED structure or area, or damage sufficient to degrade the performance of more than one SAFETY-RELATED train or more than one SAFETY-RELATED system. Degraded performance would be indicated if the affected trains and/or systems are unable to perform their intended function.

VISIBLE DAMAGE is used to differentiate between a FIRE that causes minor damage and one that has the potential to damage SAFETY-RELATED equipment. The declaration of an Alert and the activation of the Technical Support Center and Operational Support Center will provide the Emergency Director with the resources necessary to perform detailed damage assessments.

The reference to SAFETY-RELATED structures and areas excludes classification of a FIRE affecting a structure or area not associated safe operation of the plant.

Escalation of this emergency classification level, if appropriate, will be based on System Malfunctions, Fission Product Barrier Degradation or Abnormal Rad Levels / Radiological Effluent ICs.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

The site-specific list of SAFETY-RELATED structures and areas should specify those structures or areas that contain SAFETY-RELATED systems, components or functions.

HA4

Initiating Condition - ALERT

Release of a toxic, corrosive, asphyxiant or flammable gas resulting in degraded SAFETY-RELATED system performance.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas in an amount sufficient to preclude **EITHER** of the following:

- Personnel access to an area(s) containing SAFETY-RELATED equipment
- Operation of required SAFETY-RELATED equipment

AND

b. Control Room indication of degraded performance of more than one train of a SAFETY-RELATED system, or more than one SAFETY-RELATED system.

Basis:

This event represents an actual or potential substantial degradation of the level of safety of the plant in that a gas release has adversely affected the performance of multiple SAFETY-RELATED trains and/or systems; this is a precursor to a challenge to a fission product barrier.

A release of a toxic, corrosive, asphyxiant or flammable gas in certain locations and of sufficient quantity can preclude personnel access to SAFETY-RELATED equipment. In addition, a release of a corrosive or flammable gas can threaten the safe and reliable operation of plant equipment. An Alert declaration will be required if the inability to operate SAFETY-RELATED equipment due to restrictions on either personnel access or equipment operation results in degraded performance of multiple SAFETY-RELATED trains or systems.

EAL #1.a does not require atmospheric testing; it only requires the Shift Manager’s judgment that the quantity of gas released could reasonably be expected to preclude required access to, or the operability of, SAFETY-RELATED equipment. Such conditions may result in an evacuation of the affected area, imposition of access controls to the area, and/or shutting down equipment in the area to eliminate ignition sources or otherwise preclude equipment damage. This judgment may be based upon an existing job hazard analysis, report of ill effects on personnel, or operating experience with the same or similar hazards.

For EAL #1.b, “degraded performance” would be indicated if the affected trains and/or systems are unable to perform their intended function.

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.

The fact that SCBAs may be worn does not eliminate the need to declare the event.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area. This EAL does not apply to intentional inerting of containment (BWR only).

Escalation of the emergency classification level, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation or Abnormal Rad Levels / Radioactive Effluent ICs.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

HA5

Initiating Condition - ALERT

Control Room has been evacuated.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) Control Room has been evacuated.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities will be necessary.

The inability to establish plant control from outside the Control Room in a timely manner will escalate this event to a Site Area Emergency in accordance with IC HS5.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

HA6

Initiating Condition - ALERT

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 EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for an Alert.

HS1

Initiating Condition - SITE AREA EMERGENCY

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 represents an escalation in the threat to personnel and plant safety above that described by Alert IC HA1. The attack by a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA, or there has been an impact on/in the PROTECTED AREA by an AIRLINER/LARGE AIRCRAFT. Either event will require rapid assistance due to the possibility for significant and/or indeterminate damage to equipment, and casualties among the plant staff.

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.

As time and conditions allow, these events require a heightened state of readiness and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering).

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

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 within the PROTECTED AREA, 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.

Escalation of emergency classification level would be based IC HG1 after assessing the plant status during or following the attack or aircraft impact.

Developer Notes:
ECL Assignment Attributes: 3.1.3.D

HS5

Initiating Condition - SITE AREA EMERGENCY

Control Room has been evacuated and control of key safety functions has not been established.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) a. Control Room has been evacuated.

AND

b. Control of **ANY** of the following safety functions is not established from an alternate location within (site-specific number) minutes.

- Reactivity control
- Core cooling [*PWR*] / RPV water level [*BWR*]
- RCS heat removal

Basis:

This IC describes an event where control of key safety functions cannot be reestablished in a timely manner following a Control Room evacuation. The failure to promptly transfer control of the necessary SAFETY-RELATED systems 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 this emergency classification level, if appropriate, would be by Fission Product Barrier Matrix.

Developer Notes:

ECL Assignment Attributes: 3.1.3.B

The time for transfer is based on site-specific analyses as to how quickly control must be reestablished without a degradation of core cooling. This time should not be greater than 15 minutes without additional justification.

HS6

Initiating Condition - SITE AREA EMERGENCY

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 EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a Site Area Emergency.

HG1

Initiating Condition - GENERAL EMERGENCY

HOSTILE ACTION resulting in loss of key safety functions or damage to spent fuel.

Operating Mode Applicability: All

Example Emergency Action Levels:

(1) a. A HOSTILE ACTION has occurred.

AND

b. **EITHER** of the following:

1. **ANY** of the following safety functions cannot be controlled or maintained.

   - Reactivity control
   - Core cooling [*PWR*] / RPV water level [*BWR*]
   - RCS heat removal

2. Damage to spent fuel has occurred or is IMMINENT.

Basis:

This IC addresses an event in which a HOSTILE FORCE has been successful in adversely impacting the control or functionality of equipment required to maintain key safety functions. It also addresses a successful HOSTILE ACTION that results in actual or IMMINENT damage to spent fuel due either to, 1) damage to a spent fuel 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.

Developer Notes:

ECL Assignment Attributes: 3.1.4.D

HG6

Initiating Condition - GENERAL EMERGENCY

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 EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for General Emergency.

12 SYSTEM MALFUNCTION ICS/EALS

Table S-1: Recognition Category “S” Initiating Condition Matrix

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU1 Loss of offsite AC power capability to emergency busses for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA1** AC power capability to emergency busses reduced to a single power source for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS1** Loss of all offsite and all onsite AC power to emergency busses for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SG1** Prolonged loss of all offsite and all onsite AC power to emergency busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*
SU2 Plant is not brought to a required operating mode or condition within Technical Specifications LCO Action Statement Time. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA2** Automatic scram (trip) fails to shutdown the reactor. *Op. Modes: Power Operation, Startup*	**SS2** Automatic scram (trip) fails to shutdown the reactor and challenge to RCS barrier [*PWR*] or challenge to primary containment barrier [*BWR*]. *Op. Modes:* Power *Operation, Startup*	**SG2** Automatic scram (trip) fails to shutdown the reactor and extreme challenge to core cooling or RCS heat removal. *Op. Modes:* Power *Operation, Startup*
	SA3 UNPLANNED loss of SAFETY-RELATED indication in the Control Room for 15 minutes or longer with either (1) alternate indication sources not available, or (2) a significant plant transient in progress. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS3** Inability to monitor a significant plant transient in progress. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	
	SA3 [for Digital I&C only] UNPLANNED partial loss of indicating, monitoring and control functions for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS3 [for Digital I&C only]** Inability to monitor and control the plant for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU4 Fuel clad degradation greater than Technical Specification allowable limits. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*			
SU5 RCS leakage for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*			
SU6 Inadvertent criticality. *Op. Modes: Hot* Standby, *Hot Shutdown*			
		SS7 Loss of all vital DC power for 15 minutes or longer. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	

SU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Loss of offsite AC power capability to emergency busses for 15 minutes or longer.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) Loss of **ALL** offsite AC power capability to (site-specific emergency busses) for 15 minutes or longer.

Basis:

Prolonged loss of offsite power reduces required power source redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of power to AC emergency busses.

For emergency classification purposes, “capability” means that an offsite AC power source(s) is available to the site, whether or not the emergency busses are powered from it.

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

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B

At multi-unit stations, the EALs should allow credit for compensatory measures that, 1) are proceduralized and, 2) can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, “swing” generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to cross-tie AC power from an offsite power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC. These stations must also consider the impact of this condition on SAFETY-RELATED functions shared between multiple units.

SU2

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Plant is not brought to a required operating mode or condition within Technical Specifications LCO Action Statement Time.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) Plant is not brought to a required operating mode or condition within Technical Specifications LCO Action Statement Time.

Basis:

Technical Specification Limiting Conditions for Operation (LCO) define the boundaries and criteria necessary to keep the plant within its licensed design, analysis and operating envelope. If an LCO is exceeded, an associated action statement will require that some action be taken to return the plant to within its licensing basis envelope or, if that cannot be done within an allowed time frame, then an alternate action must be performed. In many cases, the alternate action is to place the unit in a different operating mode or condition within a specified time period.

The initiation of a plant shutdown required by site Technical Specifications requires a four-hour report to the NRC per 10 CFR § 50.72 (b) Non-emergency events. The plant will remain within its licensing basis envelope provided that all required mode/condition changes comply with LCO action statement times. If a Technical Specification LCO action statement requiring a mode or condition change cannot be met within the specified time frame, then there has been a potential degradation of the level of safety of the plant, and a NOUE declaration is warranted.

Some events requiring a mode change per Technical Specification may also be considered a precursor to more serious event or condition; these events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

This IC also addresses the loss radiation monitoring indications required by Technical Specifications for which a plant mode change is not specified in the applicable LCO action statement. Action statements of this type may include initiation of an alternate method of monitoring or grab sampling, closing isolation valves, placing a system in an alternate lineup or using a different system, suspending certain types of operations such as fuel movement, submittal of a special report to the commission, etc.

For purposes of assessing this IC and EAL, the determination as to whether the plant was brought to a required mode or condition within the allowable time should be made in accordance with the approach used to evaluate compliance with Technical Specifications (e.g., if the time an LCO was exceeded is earlier than the discovery time).

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B

SU4

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Fuel clad degradation greater than Technical Specification allowable limits.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) (Site-specific radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits.)

(2) (Site-specific reactor coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.)

Basis:

In this IC, the plant is outside the safety envelope defined by Technical Specifications and, as a result, is considered to be a potential degradation of the level of safety of the plant.

EAL #1 addresses site-specific radiation monitor readings that provide indication of a degradation of fuel clad integrity.

EAL #2 addresses reactor coolant samples greater than Technical Specification allowable limits for transient iodine spiking.

Escalation of this EAL to the Alert level is via the Fission Product Barrier ICs.

Developer Notes:

ECL Assignment Attributes: 3.1.1.B

For EAL #1 – Depending upon the plant design, this value may be determined using different methods (e.g., an installed radiation monitor such as a letdown system or air ejector monitor, a hand-held monitor, a remotely deployed detector, etc.). Sites are expected to use existing methods and capabilities to address this EAL.

SU5

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

RCS leakage for 15 minutes or longer.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.

(2) RCS identified leakage greater than 25 gpm for 15 minutes or longer.

Basis:

This IC may be a precursor to a more serious condition and, as a result, is considered to be a potential degradation of the level of safety of the plant. In this case, a loss of RCS mass (reactor coolant) is greater than that allowed by Technical Specifications and operators, following applicable procedures, cannot promptly isolate the leak.

These EALs should be assessed using the definitions for RCS "unidentified leakage", "pressure boundary leakage" and "identified leakage” that are contained in the plant Technical Specifications. This approach will maintain continuity between Technical Specification and EAL assessments.

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations (e.g., a mass balance calculation). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.

RCS leakage caused by the as-designed/expected operation of a relief valve does not warrant an emergency classification. For PWRs, an emergency classification would be required if the RCS leakage 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).

Escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A and 3.1.1.B

SU6

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Inadvertent criticality.

Operating Mode Applicability: Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) An UNPLANNED sustained positive period observed on nuclear instrumentation. [*BWR*]

(1) An UNPLANNED sustained positive startup rate observed on nuclear instrumentation. [*PWR*]

Basis:

This IC addresses inadvertent criticality events which can be identified using nuclear instrumentation (e.g., period monitor, startup rate monitor, etc). This IC indicates a potential degradation of the level of safety of the plant and warrants a NOUE classification.

The term “sustained” is used in order to exclude classification of expected short-term positive periods/startup rates from planned control rod movements or reactivity changes (e.g., withdrawal of a shutdown bank for a PWR). These short term positive periods/startup rates are the result of the increase in neutron population due to subcritical multiplication.

Escalation would be by the Fission Product Barrier Table, as appropriate to the operating mode at the time of the event.

Developer Notes:

ECL Assignment Attributes: 3.1.1.A

SA1

Initiating Condition - ALERT

AC power capability to emergency busses reduced to a single power source for 15 minutes or longer.

Operating Mode Applicability:	Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) AC power capability to (site-specific emergency busses) is reduced to a single power source for 15 minutes or longer.

Basis:

This IC describes a significant degradation of AC power sources (offsite and onsite) such that any additional single failure would result in a loss of all AC emergency busses. It provides an escalation path from IC SU1.

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

The subsequent loss of the remaining single power source would escalate the event to a Site Area Emergency in accordance with IC SS1.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

Some potential examples of this condition are presented below.

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

Developers should consider including site-specific examples in their IC SA1 basis.

At multi-unit stations, the EALs should allow credit for compensatory measures that 1) are proceduralized, and 2) can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, “swing” generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to cross-tie AC power from an offsite power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC. These stations must also consider the impact of this condition on SAFETY-RELATED functions shared between multiple units.

SA2

Initiating Condition - ALERT

Automatic scram (trip) fails to shutdown the reactor.

Operating Mode Applicability: Power Operation

Example Emergency Action Levels:

(1) An automatic reactor scram (trip) failed to shutdown the reactor as indicated by (site-specific indications of reactor not shutdown).

Basis:

This IC describes a failure of the reactor protection system to automatically scram (trip) the reactor following generation of an automatic scram (trip) signal. This event represents a potential substantial degradation of the level of safety of the plant in that it is a precursor event to a loss or potential loss of fission product barriers. In terms of event significance, the critical variable is how long reactor heat generation exceeds ECCS heat removal capabilities.

This IC is concerned with a failure of the Reactor Protection System to automatically scram (trip) the plant when required by design. The automatic scram (trip) signal may or may not be generated as a result of a plant transient; however, classification is required if the event results in initial post-scram (trip) conditions during which the reactor is producing more heat than the ECCS is designed to remove. This classification must be declared regardless of any subsequent actions that shutdown the reactor (e.g., a successful manual reactor trip).

Following the failure of an automatic scram (trip), operators will promptly initiate actions to shutdown the reactor. Such actions may include inserting a manual scram (trip) signal, manually driving in the control rods, emergency boration, local opening of breakers, etc. If these actions are successful, reactor heat generation will quickly fall to a level within the capabilities of the ECCS. Provided that the integrity of the RCS barrier [*PWR*], or primary containment barrier [*BWR*] is not challenged during the period of excess heat generation, there is no IMMINENT threat to any fission product barrier.

If the actions to shutdown the reactor are not successful prior to the occurrence of a challenge to the RCS barrier [*PWR*] or the primary containment barrier [*BWR*], then the event will escalate to a Site Area Emergency via IC SS2.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

For emergency classification purposes, the reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the ECCS is designed (typically 3 to 5% power). For plants using the Westinghouse CSFSTs, EALs 1.a should use the reactor power criteria associated with a Subcriticality Red Path. For BWRs, this EAL should be the APRM downscale trip setpoint.

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. A PWR that specifies a shutdown reactor power level that is less than or equal to the reactor power level that 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.

SA3

Initiating Condition - ALERT

UNPLANNED loss of SAFETY-RELATED indication in the Control Room for 15 minutes or longer with either (1) alternate indication sources not available, or (2) a significant plant transient in progress.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) a. UNPLANNED loss of greater than approximately 75% of SAFETY-RELATED indications on the main control consoles for 15 minutes or longer.

AND

b. **EITHER** of the following:

- (Site-specific alternate sources of SAFETY-RELATED indications) are also unavailable.
- A (site-specific significant plant transient) is in progress.

Basis:

This IC recognizes the degradation in plant safety arising from the loss of SAFETY-RELATED indications available on the main control consoles concurrent with the inability to obtain indications from alternate sources or the occurrence of a significant plant transient.

A "planned" loss of indications includes scheduled maintenance and testing activities.

It is assumed that if approximately 75% of the SAFETY-RELATED indications on the main control consoles are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the lost indications but, rather, use judgment to determine if this threshold is exceeded.

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

Technical Specifications will address the loss of SAFETY-RELATED indications that result from an inoperable system or component. The initiation of a Technical Specification imposed plant shutdown due to an instrument loss is reported via 10 CFR § 50.72. If the shutdown is not in compliance with Technical Specifications, a NOUE will be declared in accordance with IC SU2.

A significant plant transient is an UNPLANNED event involving one or more of the following site-specific criteria: (1) automatic [turbine runback - PWR][recirculation system flow runback - BWR] greater than 25% thermal reactor power, (2) electrical load rejection greater than [25% - PWR][site-specific MSL bypass capability - BWR] full electrical load, (3) Reactor Trip/Scram,

(4) Safety Injection Activation [PWR] Inadvertent ECCS injection [BWR], or (5) thermal power oscillations greater than 10% [BWR]

For EAL evaluation purposes, acceptable “alternate sources of SAFETY-RELATED indications" are (site-specific list of alternate indication sources).

This Alert will be escalated to a Site Area Emergency via IC SS3 if the operating crew cannot monitor a transient in progress due to a concurrent loss of alternate SAFETY-RELATED indications.

Developer Notes:

ECL Assignment Attributes: 3.1.2.B

As used here, “SAFETY-RELATED indications on the main control consoles” means the SAFETY-RELATED meters, displays, dials, gauges, readouts, status lights, etc. installed on the main control consoles from which operators determine the information necessary to operate SAFETY-RELATED systems.

Annunciators are not included in EAL #1.a because they do not provide the specific system or equipment status information, or parameter values, necessary to operate the plant, or to process through AOPs or EOPs. Compensatory measures for a loss of annunciation can be readily implemented and may include increased monitoring of main control console indications and more frequent plant rounds by non-licensed operators.

A radiation monitor indication is included in EAL #1.a if the associated radiation monitor is SAFETY-RELATED. The total population of SAFETY-RELATED indications includes any indications from SAFETY-RELATED radiation monitors. The loss of a radiation monitor indication that is not SAFETY-RELATED but important for another purpose (e.g., an EAL assessment) is addressed by other licensee processes (e.g., implementation of compensatory/contingency measures in accordance with INPO 10-007); therefore, these types of radiation monitors are not included in EAL #1.a.

Include all alternate sources of SAFETY-RELATED indications such as the plant process computer or Safety Parameter Display System.

Due to the limited number of SAFETY-RELATED systems in operation during cold shutdown, refueling, and defueled modes, no IC is indicated during these modes of operation.

This IC is not applicable to plants with SAFETY-RELATED digital I&C. These plants must use IC SA3 [*for Digital I&C Only*] – see next page.

SA3 [*for Digital I&C Only*]

Initiating Condition - ALERT

UNPLANNED partial loss of indicating, monitoring and control functions for 15 minutes or longer.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Level:

(1) UNPLANNED loss of PICS indicating, monitoring and control functions for 15 minutes or longer.

(2) UNPLANNED loss of SICS indicating, monitoring and control functions for 15 minutes or longer.

Basis:

This IC recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the indication and control systems.

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. An Alert is considered appropriate if the control room staff requires additional personnel to assist in monitoring alternative indications, manipulate equipment and restore the systems to full capability.

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

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor and control the plant.

Developer Notes:

The Process Information and Control System (PICS) is a non-safety related, augmented quality digital I&C system. It provides a screen based interface for the operators in the control room and in the remote shutdown station to control and monitor all plant parameters by interfacing with the plant automation systems. The Safety Information and Control System (SICS) is a safety related I&C system which contains both safety and non-safety related equipment. It provides the Human-System Interface (HSI) to perform control and information functions needed to monitor the plant's safety status and bring the unit to and maintain it in a safe shutdown state in case of unavailability of the PICS.

The SICS provides controls for actuating manual reactor trips and manual system level functions performed by the Protection System (PS) and the Safety Automation System (SAS) via the Priority Actuation and Control System (PACS) in order to bring the plant to and maintain it in a cold shutdown state.

Either PICS or SICS is separately capable of bringing the reactor to a safe shutdown. Therefore, a partial loss of the indicating, monitoring, and control functions when the plant has experienced the complete loss of one of the two capable systems (PICS or SICS) and a total loss of the indicating, monitoring, and control functions (i.e. inability to monitor and control the plant from the Main Control Room) is characterized by the complete loss of both capable systems (PICS and SICS).

Loss of the PICS system is indicated by no PICS terminal in the control room being functional.

SS1

Initiating Condition - SITE AREA EMERGENCY

Loss of all offsite and all onsite AC power to emergency busses for 15 minutes or longer.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency busses) for 15 minutes or longer.

Basis:

This IC is a precursor to a potential loss or loss of one or more fission product barriers. A loss of all AC emergency busses compromises all plant SAFETY-RELATED systems requiring electric power including the ECCS, shutdown cooling, containment heat removal and pressure control, and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

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

Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1.

Developer Notes:

ECL Assignment Attributes: 3.1.3.B

At multi-unit stations, the EALs should allow credit for compensatory measures that 1) are proceduralized, and 2) can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, “swing” generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to cross-tie AC power from an offsite power supply of a companion unit may take credit for the redundant power source in the associated EAL for this IC. These stations must also consider the impact of this condition on SAFETY-RELATED functions shared between multiple units.

SS2

Initiating Condition - SITE AREA EMERGENCY

Automatic scram (trip) fails to shutdown the reactor and challenge to RCS barrier [*PWR*] or challenge to primary containment barrier [*BWR*].

Operating Mode Applicability: Power Operation

Example Emergency Action Levels:

(1) a. An automatic reactor scram (trip) failed to shutdown the reactor as indicated by (site-specific indications of reactor not shutdown).

AND

b. RCS pressure reaches (site-specific lowest pressurizer PORV pressure setpoint). [*PWR*]

Suppression pool temperature reaches (site-specific Boron Injection Initiation Temperature (BIIT)). [*BWR*]

Basis:

This IC describes a failure of the reactor protection system to automatically scram (trip) the reactor, and the reactor is generating heat greater than ECCS heat removal capability for a time period long enough to challenge the integrity of the RCS barrier [*PWR*], or primary containment barrier [*BWR*].

A Site Area Emergency is warranted because:

[PWR] – RCS pressure has approached the design limits of the RCS and fuel cladding. Protection of RCS and fuel cladding integrity is now dependent upon operation of a pressure relief valve(s) until operators can shut down the reactor. There are attendant concerns including the loss of RCS mass (reactor coolant) when the pressure relief valve(s) lifts and the possibility that a valve will not fully close.

[BWR] – Exceeding the Boron Injection Initiation Temperature (BIIT) under failure to scram conditions is a fundamental indication that heat is being added to the containment at a rate that could ultimately challenge primary containment integrity. The BIIT is a function of reactor power. It is utilized to establish requirements for boron injection and deliberately lowering RPV water level following a failure-to-scram. If boron injection is initiated before suppression pool temperature reaches the BIIT, emergency RPV depressurization may be precluded at lower reactor power levels.

The BIIT is the greater of:

- The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.

- The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

Although this IC may be viewed as redundant to the fission product barrier ICs, it is included to assure timely event recognition and emergency declaration.

Escalation of this event to a General Emergency would be via IC SG2 due to prolonged power generation leading to an extreme challenge of either core cooling or RCS heat removal.

Developer Notes:

ECL Assignment Attributes: 3.1.3.B

For emergency classification purposes, the reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the ECCS is designed (typically 3 to 5% power). For plants using the Westinghouse CSFSTs, EALs 1.a should use the reactor power criteria associated with a Subcriticality Red Path. For BWRs, this EAL should be the APRM downscale trip setpoint.

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. A PWR that specifies a shutdown reactor power level that is less than or equal to the reactor power level that 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.

SS3

Initiating Condition - SITE AREA EMERGENCY

Inability to monitor a significant plant transient in progress.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) a. Loss of greater than approximately 75% of SAFETY-RELATED indications on the main control consoles.

AND

b. (Site-specific alternate sources of SAFETY-RELATED indications) are also unavailable.

AND

c. A (site-specific significant plant transient) is in progress.

Basis:

EAL #1 recognizes the threat to plant safety associated with the inability of Control Room personnel to monitor and/or control SAFETY-RELATED systems during a significant plant transient.

"Planned" and “UNPLANNED” actions are not differentiated in this IC since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

It is assumed that if approximately 75% of the SAFETY-RELATED indications on the main control consoles are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the lost indications but, rather, use judgment to determine if this threshold is exceeded.

A significant plant transient is an UNPLANNED event involving one or more of the following site-specific criteria: (1) automatic [turbine runback - PWR][recirculation system flow runback - BWR] greater than 25% thermal reactor power, (2) electrical load rejection greater than [25% - PWR][site-specific MSL bypass capability - BWR] full electrical load, (3) Reactor Trip/Scram, (4) Safety Injection Activation [PWR] Inadvertent ECCS injection [BWR], or (5) thermal power oscillations greater than 10% [BWR]

For EAL evaluation purposes, acceptable “alternate sources of SAFETY-RELATED indications" are (site-specific list of alternate indication sources).

Developer Notes:

ECL Assignment Attributes: 3.1.3.B

As used here, "SAFETY-RELATED indications on the main control consoles" means the SAFETY-RELATED meters, displays, dials, gauges, readouts, status lights, etc. installed on the main control consoles from which operators determine the information necessary to operate SAFETY-RELATED systems.

Annunciators are not included in EAL #1.a because they do not provide the specific system or equipment status information, or parameter values, necessary to operate the plant, or to process through AOPs or EOPs. Compensatory measures for a loss of annunciation can be readily implemented and may include increased monitoring of main control console indications and more frequent plant rounds by non-licensed operators.

A radiation monitor indication is included in EAL #1.a if the associated radiation monitor is SAFETY-RELATED. The total population of SAFETY-RELATED indications includes any indications from SAFETY-RELATED radiation monitors. The loss of a radiation monitor indication that is not SAFETY-RELATED but important for another purpose (e.g., an EAL assessment) is addressed by other licensee processes (e.g., implementation of compensatory/contingency measures in accordance with INPO 10-007); therefore, these types of radiation monitors are not included in EAL #1.a.

Include all alternate sources of SAFETY-RELATED indications such as the plant process computer or Safety Parameter Display System.

Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no analogous IC is included for these modes of operation.

This IC is not applicable to plants with SAFETY-RELATED digital I&C. These plants must use IC SS3 [*for Digital I&C Only*] – see next page.

SS3 [*for Digital I&C Only*]

Initiating Condition - SITE AREA EMERGENCY

Inability to monitor and control the plant for 15 minutes or longer.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Level:

(1) a. Loss of PICS for 15 minutes or longer.

AND

b. Loss of SICS for 15 minutes or longer.

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.

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

Developer Notes:

The Process Information and Control System (PICS) is a non-safety related, augmented quality digital I&C system. It provides a screen based interface for the operators in the control room and in the remote shutdown station to control and monitor all plant parameters by interfacing with the plant automation systems. The Safety Information and Control System (SICS) is a safety related I&C system which contains both safety and non-safety related equipment. It provides the Human-System Interface (HSI) to perform control and information functions needed to monitor the plant's safety status and bring the unit to and maintain it in a safe shutdown state in case of unavailability of the PICS.

The SICS provides controls for actuating manual reactor trips and manual system level functions performed by the Protection System (PS) and the Safety Automation System (SAS) via the Priority Actuation and Control System (PACS) in order to bring the plant to and maintain it in a cold shutdown state.

Either PICS or SICS is separately capable of bringing the reactor to a safe shutdown. Therefore, a partial loss of the indicating, monitoring, and control functions when the plant has experienced the complete loss of one of the two capable systems (PICS or SICS) and a total loss of the indicating, monitoring, and control functions (i.e. inability to monitor and control the plant from the MCR) is characterized by the complete loss of both capable systems (PICS and SICS).

Loss of the PICS system is indicated by no PICS terminal in the control room being functional.

SS7

Initiating Condition - SITE AREA EMERGENCY

Loss of all vital DC power for 15 minutes or longer.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) Indicated voltage is less than (site-specific bus voltage value) on **ALL** (site-specific Vital DC busses) for 15 minutes or longer.

Basis:

A total loss of DC power would compromise the ability to monitor and control plant systems. A prolonged loss of all DC power could lead to core uncovery and, ultimately, a loss of containment integrity.

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

Escalation to a General Emergency would occur via IC AG1.

Developer Notes:

ECL Assignment Attributes: 3.1.3.B

The site-specific bus voltage should be based on the minimum bus voltage necessary for the adequate operation of SAFETY-RELATED equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed. The typical minimum value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries the cell voltage is typically 1.75 Volts per cell. For a 58 string battery set the minimum voltage is typically 1.81 Volts per cell.

SG1

Initiating Condition - GENERAL EMERGENCY

Prolonged loss of all offsite and all onsite AC power to emergency busses.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency busses).

AND

b. **EITHER** of the following:

- Restoration of at least one emergency bus in less than (site-specific hours) is not likely.
- (Site-specific indication of degraded core cooling [*BWR*] / (Site-specific indication that core cooling is severely challenged [*PWR*].)

Basis:

This IC provides a General Emergency escalation path for a prolonged loss of power to all AC emergency busses. A loss of all AC emergency busses compromises all plant SAFETY-RELATED systems requiring electric power including the ECCS, shutdown cooling, containment heat removal and pressure control, and the ultimate heat sink. A prolonged loss of these busses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

<u>EAL #1.b – First bullet</u>

This EAL will prompt a General Emergency declaration prior to the end of the analyzed station blackout coping period if a power source cannot be restored by that time. Beyond this coping period, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of the degradation of multiple fission product barriers. This EAL will necessitate a declaration prior to IC FG1 being met and thus allow more time for implementation of offsite protective actions.

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, public protective actions.

Developer Notes:

ECL Assignment Attributes: 3.1.4.B

EAL #1.b – First bullet - The site-specific hours to restore AC power should be based on the station blackout coping analysis performed in accordance with 10 CFR § 50.63 and Regulatory Guide 1.155, "Station Blackout". Appropriate allowance for offsite emergency response, including evacuation of surrounding areas, should be considered. Although this IC may be viewed as redundant to the Fission Product Barrier ICs, its inclusion is necessary to better assure timely recognition and emergency declaration.

For EAL #1.b – Second bullet:

> [*BWR*] – Reactor vessel water level cannot be restored and maintained above the top of active fuel.
>
> [*PWR*] – Insert site-specific values for an incore/core exit thermocouple temperature and/or reactor vessel water level indicative of a severe challenge to core cooling. Sites may use a reactor vessel water level that corresponds to the top of active fuel and/or incore/core exit thermocouple temperatures greater than 700^{o}F. For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, use the values for the Core Cooling Orange path.

SG2

Initiating Condition - GENERAL EMERGENCY

Automatic scram (trip) fails to shutdown the reactor and extreme challenge to core cooling or RCS heat removal.

Operating Mode Applicability: Power Operation

Example Emergency Action Levels:

(1) a. An automatic reactor scram (trip) failed to shutdown the reactor as indicated by (site-specific indications of reactor not shutdown).

AND

b. **EITHER** of the following:

- (Site-specific indication that the core cooling is extremely challenged.)
- (Site-specific indication that the RCS heat removal is extremely challenged.)

Basis:

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the ECCS is designed, and efforts to bring the reactor subcritical are unsuccessful. In the event that either the core cooling and/or RCS heat removal safety functions are extremely challenged, there is an event trajectory path leading to rapid core degradation and possible melting. For this reason, the General Emergency declaration is intended to be anticipatory to that required by fission product barrier degradation; this will maximize the time available for implementation of offsite protective actions.

Developer Notes:

ECL Assignment Attributes: 3.1.4.B

For emergency classification purposes, the reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the ECCS is designed (typically 3 to 5% power). For plants using the Westinghouse CSFSTs, EALs 1.a should use the reactor power criteria associated with a Subcriticality Red Path. For BWRs, this EAL should be the APRM downscale trip setpoint.

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. A PWR that specifies a shutdown reactor power level that is less than or equal to the reactor power level that 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.

For EAL #1.b – First bullet:

[*BWR*] – Reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (as described in the EOP bases).

[*PWR*] – Insert site-specific values for an incore/core exit thermocouple temperature and/or reactor vessel water level indicative of an extreme challenge to core cooling. Sites may use a reactor vessel water level that corresponds to approximately the middle of active fuel and/or incore/core exit thermocouple temperatures greater than 1,200^{o}F. For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, use the values for the Core Cooling Red path.

For EAL #1.b – Second bullet:

[*BWR*] - Use the Heat Capacity Temperature Limit. This addresses the inability to remove heat via the main condenser and the suppression pool due to high pool water temperature.

[*PWR*] - An extreme challenge to RCS heat removal means that heat removal via the steam generators has (or soon will) become ineffective. An extreme challenge exists if the minimum level in the minimum number of steam generators cannot be maintained. Emergency (auxiliary) feedwater flow and/or steam generator level values should be determined based on the above description of the condition.

For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, use the values for the Heat Sink Red path.

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APPENDIX A - BASIS FOR RADIOLOGICAL EFFLUENT EALS

Introduction

This appendix supplements the basis information provided in Section 6 for initiating conditions AU1, AA1, AS1, and AG1.

A.1 Purpose of the Effluent ICs/EALs

ICs AU1, AA1, AS1, and AG1 provide classification thresholds for UNPLANNED and/or uncontrolled releases of radioactivity to the environment. In as much as the purpose of emergency planning at nuclear power plants is to minimize the consequences of radioactivity releases to the environment, these ICs would appear to be controlling. However, classification of emergencies on the basis of radioactivity releases is not optimum, particularly those classifications based on radiation monitor indications. Such classifications can be deficient for several reasons, including:

- In significant emergency events, a radioactivity release is seldom the initiating event, but rather, is the consequence of some other condition. Relying on an indication of a release may not be sufficiently anticipatory.
- The relationship between an effluent monitor indication caused by a release and the offsite conditions that result is a function of several parameters (e.g., meteorology, source term) which can change in value by orders of magnitude between normal and emergency conditions and from event to event. The appropriateness of these classifications is dependent on how well the parameter values assumed in pre-establishing the classification thresholds match those that are present at the time of the incident.

Section 3 of NEI 99-01 emphasizes the need for accurate assessment and classification of events, recognizing that over-classification, as well as under-classification, is to be avoided. Primary emphasis is intended to be placed on plant conditions in classifying emergency events. Effluent ICs were included, however, to provide a basis for classifying events that cannot be readily classified on the basis of plant condition alone. Plant condition ICs are included to address the precursors to radioactivity release in order to ensure anticipatory action. The effluent ICs do not stand alone, nor do the plant condition ICs. The inclusion of both categories more fully addresses the potential event spectrum and compensates for potential deficiencies in either. This is a case in which the whole is greater than the sum of the parts.

From the discussion that follows, it should become clear how the various aspects of the NEI 99-01 effluent ICs/EALs work together to provide for reasonably accurate and timely emergency classifications. During site-specific implementation of these ICs/EALs, changes to some of these aspects might appear advantageous. While site-specific changes are anticipated, caution must be used to ensure that these changes do not impact the overall effectiveness of the ICs / EALs.

A.2 Initiating Conditions

There are four radiological effluent ICs provided in NEI 99-01. The IC and the fundamental basis for the ultimate classification for the four classifications are:

NOUE (AU1) Any release of gaseous or liquid radioactivity to the environment greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.

Alert (AA1) Any release of gaseous or liquid radioactivity to the environment greater than 200 times the (site-specific effluent release controlling document) limits for 15 minutes or longer.

SAE (AS1) Actual or projected offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

GE (AG1) Actual or projected offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

For the purpose of the following bases discussion, the site-specific effluent release controlling document will be referred to as the ODCM (for those facilities that implemented Generic Letter 89-01.

The fundamental basis of AU1 and AA1 ICs differs from that for AS1 and AG1 ICs. It is important to understand the differences.

- The site-specific effluent release controlling document (Radiological Effluent Technical Specifications or the ODCM of those facilities that implemented Generic Letter 89-01) are associated with particular offsite doses and dose rate limits. For showing compliance with these limits, these documents establish methodologies for establishing effluent monitor alarm setpoints, based on defined source term and meteorology assumptions.
- AU1 and AA1 are NOT based on these particular values of offsite dose or dose rate but, rather, on the loss of plant control implied by a radiological release that exceeds a specified multiple of the ODCM release limits for a specified period of time.
- The ODCM multiples are specified only to distinguish AU1 and AA1 from non-emergency conditions and from each other. While these multiples obviously correspond to an offsite dose, the classification emphasis is on a release that does not comply with a license commitment for an extended period of time.
- While some of the example EALs for AU1 and AA1 use indications of offsite dose rates as symptoms that the ODCM may be exceeded, the IC, and the classifications, are NOT concerned with the particular value of offsite dose. While there may be quantitative inconsistencies involved with this protocol, the qualitative basis of the EAL, i.e., loss of plant control, is not affected.
- The basis of the AS1 and AG1 ICs IS a particular value of offsite dose for the event duration. AG1 is set to the value of the EPA PAG. AS1 is a fraction (10%) of the EPA PAG. As such, these ICs are consistent with the fundamental definitions of a Site Area and General Emergency.

A.3 Example Emergency Action Levels

For each of the classifications, NEI 99-01 provides some example emergency action levels and bases. Ideally, the example EALs would correspond numerically with the thresholds expressed in the respective IC. Two cases are applicable to the effluent EALs:

1. The EAL corresponds numerically to the threshold in the respective IC. For example, a field survey result of 1,000 mrem/hr for a projected condition of one hour corresponds directly to AG1.

2. The EAL corresponds numerically to the threshold in the respective IC under certain assumed conditions. For example, an effluent monitor reading that equates to 100 mrem for the projected duration of the release corresponds numerically to AS1 *if* the actual meteorology, source term, and release duration matches that used in establishing the monitor thresholds.

There are four typical example EALs:

- Effluent Monitor Readings: These EALs are pre-calculated values that correspond to the condition identified in the IC for a given set of assumptions (AU1 and AA1 only).
- Field Survey Results: These example EALs are included to provide a means to address classifications based on results from field surveys.
- Perimeter Monitor Indications: For sites having them, perimeter monitors can provide a direct indication of the offsite consequences of a release.
- Dose Assessment Results: These example EALs are included to provide a means to address classifications based on dose assessments (AS1 and AG1 only).

A.3.1 Effluent Monitor Readings

AU1 and AA1

ODCMs provide a methodology for determining default and batch-specific effluent monitor alarm setpoints. These setpoints are intended to show that releases are within . The applicable limits are typically 500 mrem/year whole body or 3000 mrem/year skin from noble gases. (Inhalation dose rate limits are not addressed here since the specified surveillance involves collection and analysis of composite samples. This after-the-fact assessment could not be made in a timely manner conducive to accident classification.) These setpoints are calculated using default source terms or batch-specific sample isotopic results and annual average χ/Q. Since the meteorology data is pre-defined, there is a direct correlation between the monitor setpoints and the ODCM limits. Although the actual χ/Q may be different, NUREG-1022, *Event Reporting Guidelines 10 CFR* § *50.72 and* § *50.73*, provided *"..Annual average meteorological data should be used for determining offsite airborne concentrations of radioactivity to maintain consistency with the Technical Specifications (TS) for reportability thresholds.*" The ODCM methodology is based on long term continuous releases. However, its use here in a short term release situation is appropriate. Remember that the AU1 and AA1 ICs are based on a loss of plant control indicated by the failure to comply with a multiple of the ODCM release limits for an extended period and that the ODCM provides the methodology for showing compliance with the ODCM limits.

To obtain the thresholds, multiply the ODCM setpoint for each monitor by 2 (AU1) or 200 (AA1). It would be preferable to reference "*2 x ODCM setpoint"* or *"200 x ODCM setpoint"* as the threshold. In this manner, the EAL would always change in step with changes in the ODCM setpoint (e.g., for a batch or special release). In actual practice, there may be an "warning" and a "high" alarm setpoint. The setpoint that is closest in value to the ODCM limit should be used. Facility ODCMs may lower the actual setpoint to provide an administrative "safety margin".

Also, if there is more than one unit or release stack on the site, the ODCM limits may be apportioned. Two possible approaches to obtain the thresholds are:

- The "2x" and "200x" multiples could be increased to address the reduced setpoints. For example, if the stack monitor was set to 50% of the ODCM limit, the threshold could be set to "4x" and "400x" the setpoint on that monitor.
- The reduced setpoints could be ignored and the "2x" and "200x" multiples used as specified. While numerically conservative, using a single set of multipliers would probably be desirable from a human engineering standpoint.

While assessments with real meteorology may have provided a basis for escalating to AS1 (or AG1), the assessments could not confirm the AU1 or AA1 classifications since compliance with the ODCM is demonstrated using *annual average* meteorology – not actual meteorology.

Nonetheless, dose assessments are important components of the overall accident assessment activities when significant radioactivity releases have occurred or are projected. Dose assessment results, when they become available, may indicate that an escalation to a higher classification is necessary. AS1 and AG1 both provide that, if dose assessment results are available, the classification should be based on the dose assessment result rather than the effluent radiation monitor EAL.

In typical practice, the radiological effluent monitor alarms would have been set, on the basis of ODCM requirements, to indicate a release that could exceed the ODCM limits. Alarm response procedures call for an assessment of the alarm to determine whether or not ODCM limits have been exceeded. Utilities typically have methods for rapidly assessing an abnormal release in order to determine whether or not the situation is reportable under 10 CFR § 50.72. Since a radioactivity release of a magnitude comparable to the ODCM limits will not create a need for offsite protective measures, it would be reasonable to use these abnormal release assessment methods to initiate dose assessment techniques using actual meteorology and projected source term and release duration.

<u>AS1 and AG1</u>

Classifications should be made under these EALs based on dose assessments with real meteorology. Dose assessments are important components of the overall accident assessment activities when significant radioactivity releases have occurred or are projected. AS1 and AG1 both provide the classification based on the basis of real-time dose assessment results rather than a default effluent radiation monitor EAL as provided in AU1 and AA1.

A.3.2 Perimeter Monitor, Field Survey Results, Dose Projection Results

<u>AU1 and AA1</u>

As discussed previously, the threshold in these ICs is based on exceeding a multiple of the ODCM release limits for an extended period. The applicable ODCM limit is the instantaneous dose rate provided in Standard Technical Specification (STS) 3.11.2.1. While these three EALs are also expressed in dose rate, they are dependent on *actual* meteorology. However, compliance with the ODCM is demonstrated using *annual average* meteorology. Due to this, the only time that there would be a 1:1 correlation between the IC and these EALs is when the value of the actual meteorology matched the annual average - an unlikely situation. For this reason, these EALs can only be indirect indicators that the ODCM may be exceeded. The three example EALs are consistent with the fundamental basis of AU1 and AA1, that of an uncontrolled radioactivity release that indicates a loss of plant control. A dose rate, at or beyond the site boundary, greater than 0.1 mR/hr for 60 minutes or 10.0 mR/hr for 15 minutes is consistent with this fundamental basis, regardless of the lack of numerical correlation to the ODCM. The time periods chosen for the NOUE AU1 (60 minutes) and Alert AA1 (15 minutes) are indicative of the relative risks based on the loss of ability to terminate a release.

The numeric values shown in AU1 and AA1 are based on a release rate not exceeding 500 mrem per year, converted to a rate of: $500 \div 8{,}766 = 0.057$ mR/hr. If we take a multiple of 2, as specified in the NOUE threshold, this equates to a dose rate of about 0.11 mR/hr, which rounds to the 0.1 mR/hr specified in AU1. Similarly for the AA1 EALs, we obtain 10 mR/hr.

In AU1 and AA1, reference is made to *automatic real-time dose assessment capability*. In AS1 and AG1, the reference is to *dose assessment*. This distinction was made since it is unlikely that a dose assessment using manual methods would be initiated without some prior indication.

AS1 and AG1

The perimeter monitor and field survey results are included to provide a means for classification based on actual measurements. There is a 1:1 correlation (with consideration of release duration) between these EALs and the IC since all are dependent on actual meteorology.

Dose projection result EALs are included to provide a basis for classification based on results from assessments triggered at lower emergency classifications.

Although the IC references TEDE and thyroid CDE as criteria, field survey results and perimeter monitor indications will generally not be reported in these dose quantities, but rather in terms of a dose rate. For this reason, the field survey EALs are based on a β-γ dose rate and a thyroid CDE value, both assuming one hour of exposure (or inhalation). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used for the field survey and/or perimeter monitor EALs.

A.4 Interface Between ODCM and ICs/EALs

For AU1 and AA1, a strong link was established with the facility's ODCM. It was the intent of the NEI 99-01 EAL Task Force to have the AU1 and AA1 EALs indexed to the ODCM alarm setpoints. This was done for several reasons:

- To allow the EALs to use the monitor setpoints already in place in the facility ODCM, thus eliminating the need for a second set of values as the EALs. The EAL could reference "2x ODCM Setpoint" or "200x ODCM Setpoint" for the monitors addressed in the ODCM. Extensive calculations would only be necessary for monitors not addressed in the ODCM.
- To take advantage of the alarm setpoint calculational methodology already documented in the facility ODCM.

- To ensure that the operators had an alarm to indicate the abnormal condition. If the monitor threshold was less than the default ODCM setpoint, the operators could be in the position of having exceeded an EAL and not knowing it.
- To simplify the IC/EAL by eliminating the need to address planned and UNPLANNED releases, continuous or batch releases, monitored or unmonitored releases. Any release that complies with the Radiological Effluent Technical Specifications (RETS) (or ODCM controls for utilities that have implemented GL 89-01) would not exceed a monitor threshold.
- To eliminate the possibility of a planned release (e.g., containment / primary containment purge) resulting in effluent radiation monitor readings that exceed a classification threshold that was based on a different calculation method. ODCMs typically require specific alarm setpoints for such releases. If the release can be authorized under the provisions of the ODCM/RETS, an emergency classification is not warranted. If the monitor threshold is indexed to the ODCM setpoint (e.g., "...2 x ODCM setpoint...") the monitor EAL will always change in step with the ODCM setpoint.
- Although the ODCM addresses long term routine releases, its use here for short term releases is appropriate. The IC is specified in terms of a release that exceeds ODCM for an extended period of time. Compliance to the ODCM is shown using the ODCM methodology.

A.5 Setpoints versus Monitor EALs

Effluent monitors typically have provision for two separate alarm setpoints associated with the level of measured radioactivity. (There may be other alarms for parameters such as low sample flow.) These setpoints are typically established by the facility ODCM. As such, at most sites the values of the monitor thresholds will not be implemented as actual alarm setpoints, but would be tabulated in the classification procedure. If the monitor thresholds are calculated as suggested herein they will be higher than the ODCM alarm setpoints by at least a factor of two (i.e., AU1). This alarm alerts the operator to compare the monitor indication to the thresholds. The NEI 99-01 effluent EALs do NOT require alarm setpoints based on the monitor EALs. However, if spare alarm channels are available (e.g., high range channels), the monitor threshold could be used as the alarm setpoint.

A.6 The Impact of Source Term

The ODCM methodology should be used for establishing the monitor thresholds for ICs AU1 and AA1. The ODCM provides a default source term based on expected releases. In many cases, the ODCM source term is derived from expected and/or design releases tabulated in the FSAR.

APPENDIX B - BASIS FOR PERMANENTLY DEFUELED STATION EALS

Recognition Category D was written to provide a stand-alone set of IC/EALs for Permanently Defueled Stations. IC/EALs from Recognition Category A, C, F, S, and H were reviewed and where applicable have been included to address all Permanently Defueled station events.

A Permanently Defueled station is basically a spent fuel storage facility. This appendix is based on the assumption that the spent fuel was generated by an operating nuclear power station under a 10 CFR § 50 license that has ceased operations and intends to store the spent fuel for some period of time. The spent fuel is stored in a pool of water that serves as both the cooling medium for decay heat and shielding from direct radiation. The primary functions of this pool configuration become the emphasis of emergency classification methodology.

When in the permanently defueled condition, the licensee receives approval for exemption from specific emergency planning requirements. These exemptions must be approved by the NRC. The source term and relative risks associated with pool storage are the basis for maintaining only an onsite emergency plan. Calculations are provided in the licensing process that quantify radioactive releases associated with plausible accidents as documented in the stations Safety Analysis Report (SAR).

The emergency classification levels used are those provided by NUREG-0654/FEMA–REP-1. The NOUE emergency classification levels provide an increased awareness for abnormal conditions. The Alert emergency classification levels are specific to the actual or potential effects on the spent fuel in storage. The source term and motive force available in the permanently defueled condition is insufficient to warrant Site Area Emergency or General Emergency classification levels. Analyses for the credible design basis accidents are provided in the SAR.

Section 3 of NEI 99-01 emphasizes the need for accurate assessment and classification of events, recognizing that over-classification, as well as under-classification, is to be avoided. Primary emphasis is intended to be placed on observable conditions in classifying emergency events. In the permanently defueled condition, these conditions are primarily associated with the spent fuel, the spent fuel pool systems used to provide cooling, and shielding. Effluent IC/EALs were included, however, to provide a basis for classifying events that cannot be readily classified based on an observable event or condition alone.

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APPENDIX C - ACRONYMS AND ABBREVIATIONS

AC .... Alternating Current
AOP .... Abnormal Operating Procedure
APRM .... Average Power Range Meter
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
CTMT/CNMT .... Containment
CSF .... Critical Safety Function
CSFST .... Critical Safety Function Status Tree
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
EPIP .... Emergency Plan Implementing Procedure
EPR .... Evolutionary Power Reactor
EPRI .... Electric Power Research Institute
ERG .... Emergency Response Guideline
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GE .... General Emergency
HCTL .... Heat Capacity Temperature Limit
HPCI .... High Pressure Coolant Injection
HSI .... Human System Interface
IC .... Initiating Condition
ID .... Inside Diameter
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
LOCA .... Loss of Coolant Accident
MCR .... Main Control Room
MSIV .... Main Steam Isolation Valve
MSL .... Main Steam Line
mR, mRem, mrem, mREM .... milliRoentgen
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
NOUE .... Notification Of Unusual Event
NUMARC .... 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
PICS .... Process Information and Control System
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PS .... Protection System
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RCC .... Reactor Control Console
RCIC .... Reactor Core Isolation Cooling
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RVLIS .... Reactor Vessel Level Indicating System
RWCU .... Reactor Water Cleanup
SAR .... Safety Analysis Report
SAS .... Safety Automation System
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SG .... Steam Generator
SI .... Safety Injection
SICS .... Safety Information and Control System
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
WOG .... Westinghouse Owners Group

APPENDIX D - DEFINITIONS

Selected words in NEI 99-01 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.

AFFECTING NORMAL PLANT OPERATIONS: Events that result in a change to the current reactor power level or entry into an emergency operating procedure.

AIRLINER/LARGE AIRCRAFT: Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition). General aviation aircraft such as a Cessna, Piper and Learjet-type private plane, or a helicopter, are NOT considered to be an AIRLINER OR LARGE AIRCRAFT.

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

CONTAINMENT CLOSURE: (Insert a site-specific definition for this term.) **Developer Note –** The procedurally defined conditions or actions taken to secure containment (primary or secondary for BWR) and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

EXPLOSION: A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to damage permanent structures, systems, or components. A release of steam from a steamline (or a pressurized high temperature water line) is not, in and of itself, evidence of an explosion; this determination should based on the failure mode and the resulting damage to surrounding systems, structures and components.

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

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

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 regardless of anticipated or in-progress mitigation or corrective actions. Where IMMINENT timeframes are specified, they shall apply.

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.

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.)

SAFETY-RELATED: A system, structure or component relied upon to remain functional during and following a design basis event in order to protect the integrity of the reactor coolant pressure boundary; shut down the reactor and maintain it in a safe shutdown condition; or prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in §50.34(a)(1) or §100.11.

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 process or ventilation system line that cannot be isolated (closed), 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.

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

VISIBLE DAMAGE: Damage to a SAFETY-RELATED structure of sufficient visual impact to cause concern about the structure's integrity or ability to perform it's intended design function, or concern for the operability or reliability of systems or components within the structure. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.