

**Methodology for Development  
of Emergency Action Levels  
Advanced Passive Light Water Reactors**

July 2009

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## **FOREWORD**

The initial version of this document was written based on NEI 99-01, Revision 5. The NEI EAL Task Force coordinated with the passive light water reactor vendors to consider each IC/EAL and determine its applicability to the design of the plants and to determine what additional IC/EALs would be required. Those ICs/EALs not applicable due to the design were not included.

The approved Design Certification does not include detailed design data for those items specific to a site location. In many cases this data is necessary to determine EAL thresholds. In these cases this document provides a [Site specific] placeholder.

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## EXECUTIVE SUMMARY

NEI 07-01, Methodology for Development of Emergency Action Levels, Advanced Passive Light Water Reactors, is based on the EAL work accomplished through the development of NUMARC/NESP 007, NEI 99-01 Methodology for Development of Emergency Action Levels Revision 4 and Revision 5.

The EAL Task Force identified eight characteristics that were to be incorporated into model EALs. Experience to date has shown these considerations to be valid. These were:

- (1) Consistency (i.e., the EALs would lead to similar decisions under similar circumstances at different plants);
- (2) Human engineering and user friendliness;
- (3) Potential for classification upgrade only when there is an increasing threat to public health and safety;
- (4) Ease of upgrading and downgrading;
- (5) Thoroughness in addressing, and disposing of, the issues of completeness and accuracy raised regarding NUREG-0654, Appendix 1;
- (6) Technical completeness and appropriateness for each classification level;
- (7) A logical progression in classification for combinations of multiple events;
- (8) Objective, observable values.

Based on the information gathered and reviewed, the Task Force has developed generic EAL guidance. Because of the wide variety of presentation methods (formats) used at different utilities, the Task Force believes that specifying guidance as to what each IC and EAL should address, and including sufficient basis information for each EAL will best assure uniformity of approach. The information is presented by Recognition Category:

- A - Abnormal Rad Levels/Radiological Effluent
- C - Cold Shutdown / Refueling System Malfunction
- E - Events Related to Independent Spent Fuel Storage Installations
- F - Fission Product Barrier Degradation
- H - HAZARDS or OTHER Conditions Affecting Plant Safety
- S - System Malfunction

Note: Category D for permanently defueled stations is not included in NEI 07-01. When, in the future, this condition becomes viable a set of EALs for the permanently defueled units must be developed.

Each of the EAL guides in the Recognition Categories is structured in the following way:

- Recognition Category - As described above.
- Emergency Class – Notification of Unusual Event (NOUE), Alert, Site Area Emergency or General Emergency.
- Initiating Condition - Symptom- or Event-Based, Generic Identification and Title.
- Operating Mode Applicability - Power Operation, Hot Standby, Safe/Stable Shutdown, Cold Shutdown, Refueling, Defueled, All, or Not Applicable.
- Example Emergency Action Level(s) corresponding to the IC.
- Basis information for plant-specific readings and factors that may relate to changing the generic IC or EAL to a different emergency class, such as for Loss of All AC Power.

For Recognition Category F, the EAL information is presented in a matrix format. The presentation method was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. For Recognition Category F, the EALs are arranged by safety function or fission product

barrier. Classifications are based on various combinations of safety function or fission product barrier challenges.

The EAL Guidance has the primary threshold for NOUE as operation outside the safety envelope for the plant as defined by plant Technical Specifications, including LCOs and Action Statement Times. In addition, certain precursors of more serious events such as earthquakes are included in NOUE EALs. This provides a clear demarcation between the lowest emergency class and "non-emergency" notifications specified by 10 CFR 50.72.

## ACRONYMS

AC/ac	Alternating Current
ADS	Automatic Depressurization System
AP1000	Advanced Passive 1000 Mw PWR
APRM	Average Power Range Monitor
ARM	Area Radiation Monitor
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CAS	Central Alarm Station
CDE	Committed Dose Equivalent
CET	Core Exit Thermocouple
CFR	Code of Federal Regulations
Ci	Curie
CMT/CNMT	Containment
CSF	Critical Safety Function
CSFST	Critical Safety Function Status Tree
DAS	Diverse Actuation System
DC/dc	Direct Current
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EFS	Communication System
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPG	Emergency Procedure Guideline
EPIP	Emergency Plan Implementing Procedure
EPRI	Electric Power Research Institute
ERG	Emergency Response Guideline
ESBWR	Economic Simplified Boiling Water Reactor
FAA	Federal Aviation Administration
FAPCS	Fuel and Auxiliary Pools Cooling System
FAQ	Frequently Asked Question
FBI	Federal Bureau of Investigation
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GE	General Emergency
HCTL	Heat Capacity Temperature Limit
IC	Initiating Condition
IRWST	In Containment Refueling Water Storage Tank
ISFSI	Independent Spent Fuel Storage Installation
$K_{eff}$	Effective Neutron Multiplication Factor
LCO	Limiting Condition of Operation
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MSL	Main Steam Line
MSIV	Main Steam Isolation Valve
mR	milliRoentgen
Mw	Megawatt
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NOUE	Notification Of Unusual Event
NPP	Nuclear Power Plant
OBE	Operating Basis Earthquake



## ACRONYMS

OCA	Owner Controlled Area
ODCM	Off-site Dose Calculation Manual
ORO	Offsite Response Organization
PA	Protected Area
PAG	Protective Action Guideline
PCS	Primary Containment System
PGA	Peak Ground Acceleration
PIP	Plant Investment Protection
PLS	Plant Control System
PMS	Plant Monitoring and Control System (AP1000)
PMS	Pool Monitoring System (ESBWR)
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
psig	Pounds per Square Inch Gauge
Q-DCIS	Safety Related Distributed Control and Information System
R	Rem
RCS	Reactor Coolant System
RMS	Radiation Monitoring System
RNS	Normal Residual Heat Removal System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling System
SAG	Severe Accident Guideline
SCBA	Self Contained Breathing Apparatus
SBGTS	Stand-By Gas Treatment System
SG	Steam Generator
SPDS	Safety Parameter Display System
SRNM	Source Range Neutron Monitor (AP1000)
SRNM	Startup Range Neutron Monitor (ESBWR)
SRO	Senior Reactor Operator
SSE	Safe Shutdown Earthquake
TEDE	Total Effective Dose Equivalent
TOAF/TAF	Top of Active Fuel
TS	Technical Specification
TSC	Technical Support Center
TVS	Closed Circuit Television System (AP1000)

## 1.0 METHODOLOGY FOR DEVELOPMENT OF EMERGENCY ACTION LEVELS

### 1.1 Background

NEI 07-01, Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors, is based on the EAL work accomplished through the NUMARC/NESP 007, NEI 99-01 Revision 4 and Revision 5 development process.

Nuclear utilities must respond to a formal set of threshold conditions that require plant personnel to take specific actions with regard to notifying state and local governments and the public when certain off-normal indicators or events are recognized. Emergency classification levels are defined in 10 CFR 50. Levels of response and the conditions leading to those responses are defined in joint NRC/FEMA guidelines contained in Appendix 1 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.

The EAL Task Force developed a systematic approach and supporting basis for EAL development. This methodology developed a set of generic EAL guidelines, together with the basis for each, such that they could be used and adapted by each utility on a consistent basis. The review of the industry's experiences with EALs, in conjunction with regulatory considerations, was applied directly to the development of this generic set of EAL guidelines. The generic guidelines were intended to clearly define conditions that represent increasing risk to the public and can give consistent classifications when applied at different sites. The guidance presented here is not intended to be applied to plants as-is. It is intended to give the user the logic for developing site specific EALs (i.e., instrument readings, etc.) using site specific EAL presentation methods (formats). Basis information is provided to aid station personnel in preparation of their own site specific EALs, to provide necessary information for training, and for explanation to state and local officials. In addition, state and local requirements have not been reflected in the generic guidance and should be considered on a case-by-case basis with appropriate state and local emergency response organizations.

It is important that the NEI EALs be treated as an integrated package. Selecting only portions of this guidance for use in developing site specific EALs could lead to inconsistent or incomplete EALs unless explicitly allowed.

Although the basic concerns with barrier integrity and the major safety problems of nuclear power plants are similar across plant types, design differences will have a substantial effect on EALs. In these cases, EAL guidelines unique to AP1000 and ESBWR are specified. These passive design plants incorporate the requirements contained in EPRI Advanced Light Water Reactor (ALWR) Utilities Requirements Document Volume 3, ALWR Passive Plant. Accordingly, many of the plant safety features for both designs are functionally equivalent.

The original EAL Task Force identified eight characteristics that were to be incorporated into model EALs. Experience to date has shown these considerations to be VALID. These were:

- (1) Consistency (i.e., the EALs would lead to similar decisions under similar circumstances at different plants);
- (2) Human engineering and user friendliness;
- (3) Potential for classification upgrade only when there is an increasing threat to public health and safety;
- (4) Ease of upgrading and downgrading;
- (5) Thoroughness in addressing, and disposing of, the issues of completeness and accuracy raised regarding NUREG-0654, Appendix 1;
- (6) Technical completeness and appropriateness for each classification level;
- (7) A logical progression in classification for combinations of multiple events;
- (8) Objective, observable values.

Based on the information gathered and reviewed, the Task Force has developed generic EAL guidance. Because of the wide variety of presentation methods (formats) used at different utilities, the Task Force believes that specifying guidance as to what each IC and EAL should address, and including sufficient basis information for each EAL will best assure uniformity of approach. The information is presented by Recognition Category:

- A - Abnormal Rad Levels/Radiological Effluent
- C - Cold Shutdown / Refueling System Malfunction
- E - Independent Spent Fuel Storage Installations (ISFSI)
- F - Fission Product Barrier
- H - Hazards and Other Conditions Affecting Plant Safety
- S - System Malfunction

Each of the EAL guides in Recognition Categories A, C, E, H, and S is structured in the following way:

- Recognition Category - As described above.
- Emergency Classification Levels - NOUE, Alert, Site Area Emergency or General Emergency.
- Initiating Condition - Symptom- or Event-Based, Generic Identification and Title.
- Operating Mode Applicability - Power Operation, Hot Standby, Hot Shutdown, Cold Shutdown, Refueling, Defueled, All, or Not Applicable.
- Example Emergency Action Level(s) corresponding to the IC.
- Basis information for plant specific readings and factors that may relate to changing the generic IC or EAL to a different emergency classification level, such as for Loss of All AC Power.
- EAL developer information – Information used to aid licensees in the development of site specific EALs.

For Recognition Category F, the EAL information is presented in a matrix format. The presentation method was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. For category F, the EALs are arranged by safety function, or fission product barrier. Classifications are based on various combinations of function or barrier challenges.

The EAL Guidance has the primary threshold for NOUE as operation outside the safety envelope for the plant as defined by plant Technical Specifications, including LCOs and Action Statement Times. In addition, certain precursors of more serious events such as loss of offsite AC power and earthquakes are included in NOUE EALs. This provides a clear demarcation between the lowest emergency classification level and "non-emergency" notifications specified by 10 CFR 50.72.

In 2006, the nuclear power revival of new plants with the advanced passive designs was being planned. The NEI EAL Task Force developed this document to address the Westinghouse AP1000 and the GE Hitachi Nuclear Energy ESBWR designs. LWR designs which use digital instrumentation and control systems may use CU7, CA7, SA7, and SS7 as a template for EAL development.

## 2.0 CHANGES INCORPORATED WITH NEI 07-01

Changes will be identified in this section for future revisions.

### 3.0 DEVELOPMENT OF BASIS FOR GENERIC APPROACH

The generic guidance provided in this document addresses radiological emergency preparedness. Non-radiological events are included in the classification scheme only to the extent that these events represent challenges to the continued safety of the reactor plant and its operators. There are existing reporting requirements (Environmental Protection Agency, Occupational Safety and Health Administration) under which utilities operate. There are also requirements for emergency preparedness involving hazardous chemical releases. While the proposed classification structure could be expanded to include these non-radiological hazards, these events are beyond the scope of this document.

This classification scheme is based on the four classification levels promulgated by the NRC as the standard for the United States. The NRC has determined that domestic nuclear facilities would continue to classify events using the four classification levels and that the NRC would re-classify the event in any international communication.

#### 3.1 Definitions Used in Developing EAL Methodology

Based on the above review of regulations, review of common utility usage of terms, discussions among Task Force members, and existing published information, the following definitions apply to the generic EAL methodology:

**EMERGENCY CLASS:** One of a minimum set of names or titles, established by the Nuclear Regulatory Commission (NRC), for grouping off-normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time-sensitive on-site and off-site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called:

- Notification of Unusual Event
- Alert
- Site Area Emergency
- General Emergency

**INITIATING CONDITION (IC):** One of a predetermined subset of nuclear power plant conditions where either the potential exists for a radiological emergency, or such an emergency has occurred.

**Discussion:**

In NUREG-0654, the NRC introduced, but did 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 utilities.

Defined in this manner, an IC is an emergency condition which sets it apart from the broad class of conditions that may or may not have the potential to escalate into a radiological emergency. It can be a continuous, measurable function that is outside Technical Specifications, such as elevated RCS temperature or falling reactor coolant level (a symptom). It also encompasses occurrences such as FIRE (an event) or reactor coolant pipe failure (an event or a barrier breach).

**EMERGENCY ACTION LEVEL (EAL):** A pre-determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (on-site or off-site); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

## **Discussion:**

The term "emergency action level" has been defined by example in the regulations, as noted in the above discussion concerning regulatory background. The term had not, however, been defined operationally in a manner to address all contingencies.

There are times when an EAL will be a threshold point on a measurable continuous function, such as a primary system coolant leak that has exceeded Technical Specifications for a specific plant.

At other times, the EAL and the IC will coincide, both identified by a discrete event that places the plant in a particular emergency class. For example, "Train Derailment On-site" is an example of an "NOUE" IC in NUREG-0654 that also can be an event-based EAL.

### 3.2 Perspective

The purpose of this effort is to define a methodology for EAL development that will better assure a consistent emergency classification commensurate with the level of risk. The approach must be easily understood and applied by the individuals responsible for on-site and off-site emergency preparedness and response. In order to achieve consistent application, this recommended methodology must be accepted at all levels of application (e.g., licensed operators, health physics personnel, facility managers, off-site emergency agencies, NRC and FEMA response organizations, etc.).

Commercial nuclear facilities are faced with a range of public service and public acceptance pressures. It is of utmost importance that emergency regulations be based on as accurate an assessment of the risk as possible. There are evident risks to health and safety in understating the potential hazard from an event. However, there are both risks and costs to alerting the public to an emergency that exceeds the true threat. This is true at all levels, but particularly if evacuation is recommended.

### 3.3 Recognition Categories

ICs and EALs can be grouped in one of several schemes. This generic classification scheme incorporates symptom-based, event-based, and barrier-based ICs and EALs.

The symptom-based category for ICs and EALs refers to those indicators that are measurable over some continuous spectrum, such as core temperature, coolant levels, containment pressure, etc. When one or more of these indicators begin to show off-normal readings, reactor operators are trained to identify the probable causes and potential consequences of these "symptoms" and take corrective action. The level of seriousness indicated by these symptoms depends on the degree to which they have exceeded Technical Specifications, the other symptoms or events that are occurring contemporaneously, and the capability of the licensed operators to gain control and bring the indicator back to safe levels.

Event-based EALs and ICs refer to occurrences with potential safety significance. The range of seriousness of these "events" is dependent on the location, number of contemporaneous events, remaining plant safety margin, etc.

Barrier-based EALs and ICs refer to the level of challenge to principal barriers used to assure containment of radioactive materials contained within a nuclear power plant. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers concurrently under challenge. In reality, barrier-based EALs are a subset of symptom-based EALs that deal with symptoms indicating fission product barrier challenges. These barrier-based EALs are primarily derived from Emergency Operating Procedures (EOPs) Critical Safety

Function (CSF) Status Tree Monitoring for the AP1000 and Emergency Procedure and Severe Accident Guidelines (EPGs/SAGs) for the ESBWR. Challenge to one or more barriers generally is initially identified through instrument readings and periodic sampling. Under present barrier-based EALs, deterioration of the reactor coolant system pressure boundary or the fuel clad barrier usually indicates an Alert condition, two barriers under challenge a Site Area Emergency, and loss of two barriers with the third barrier under challenge is a General Emergency. The fission product barrier matrix described in Section 5-F is a hybrid approach that recognizes that some events may represent a challenge to more than one barrier, and that the containment barrier is weighted less than the reactor coolant system pressure boundary and the fuel clad barriers.

Symptom-based ICs and EALs are most easily identified when the plant is in a normal startup, operating or safe/stable shutdown mode of operation, with all of the barriers in place and the plant's instrumentation and emergency safeguards features fully operational as required by Technical Specifications. It is under these circumstances that the operations staff has the most direct information of the plant's systems, displayed in the main Control Room. As the plant moves through the decay heat removal process toward cold shutdown and refueling, barriers to fission products are reduced (i.e., reactor coolant system pressure boundary may be open) and fewer of the safety systems required for power operation are required to be fully operational. Under these plant operating modes, the identification of an IC in the plant's operating and safety systems becomes more event based, as the instrumentation to detect symptoms of a developing problem may not be fully effective; and systems are partially disabled as permitted by the plant's Technical Specifications.

Barrier-based ICs and EALs also are heavily dependent on the ability to monitor instruments that indicate the condition of plant operating and safety systems. Fuel cladding integrity and reactor coolant levels can be monitored through several indicators when the plant is in a normal operating mode, but this capability is much more limited when the plant is in a refueling mode, when many of these indicators are disconnected or off-scale. The need for this instrumentation is lessened, however, and alternate instrumentation is placed in service when the plant is shut down.

It is important to note that in some operating modes there may not be definitive and unambiguous indicators of containment integrity available to Control Room personnel. For this reason, barrier-based EALs should not place undue reliance on assessments of containment integrity in all operating modes. Generally, Technical Specifications relax maintaining containment integrity requirements in cold shutdown and refueling in order to provide flexibility in performance of specific tasks during shutdown conditions. Containment pressure and temperature indications may not increase if there is a pre-existing breach of containment integrity.

Several categories of emergencies have no instrumentation to indicate a developing problem, or the event may be identified before any other indications are recognized. A reactor coolant pipe could break; FIRE alarms could sound; radioactive materials could be released; and any number of other events can occur that would place the plant in an emergency condition with little warning. For emergencies related to the reactor system and safety systems, the ICs shift to an event based scheme as the plant mode moves toward cold shutdown and refueling modes. For non-radiological events, such as FIRE, external floods, wind loads, etc., as described in NUREG-0654 Appendix 1, event-based ICs are the norm.

In many cases, a combination of symptom-, event- and barrier-based ICs will be present as an emergency develops. In a loss of coolant accident (LOCA), for example:

- Coolant level is dropping; (symptom)
- There is a leak of some magnitude in the system (pipe break, safety valve stuck open) that exceeds plant capabilities to make up the loss; (barrier breach or event)
- Core (coolant) temperature is rising; (symptom) and
- At some level, fuel failure begins with indicators such as high off-gas, high coolant activity samples, etc. (barrier breach or symptom)

### 3.4 Design Differences

Although the same basic concerns with barrier integrity and the major safety problems of nuclear power plants are similar, design differences will have a substantial effect on EALs. In these cases, EAL guidelines unique to AP1000 and ESBWR are specified. These passive design plants incorporate requirements contained in EPRI Advanced Light Water Reactor Utilities Requirements Document Volume 3, ALWR Passive Plant. Accordingly, many of the plant safety features for both designs are functionally equivalent.

### 3.5 Required Characteristics

Eight characteristics that are incorporated into model EALs are identified below:

- (1) Consistency (i.e., the EALs would lead to similar decisions under similar circumstances at different plants);
- (2) Human engineering and user friendliness;
- (3) Potential for classification upgrade only when there is an increasing threat to public health and safety;
- (4) Ease of upgrading and downgrading;
- (5) Thoroughness in addressing, and disposing of, the issues of completeness and accuracy raised regarding NUREG-0654 Appendix 1;
- (6) Technical completeness for each classification level;
- (7) A logical progression in classification for multiple events; and
- (8) Objective, observable values.

The EAL development methodology pays careful attention to these eight characteristics to assure that all are addressed in the proposed EALs. The most pervasive and complex of the eight is the first—"consistency." The common denominator that is most appropriate for measuring consistency among ICs and EALs is relative risk. The approach taken in the development of these EALs is based on risk assessment to set the boundaries of the emergency classes and assure that all EALs that trigger that emergency class are in the same range of relative risk. Precursor conditions of more serious emergencies also represent a potential risk to the public and must be appropriately classified.

### 3.6 Emergency Class Descriptions

There are three considerations related to emergency classes. These are:

- (1) The potential impact on radiological safety, either as now known or as can be reasonably projected;
- (2) How far the plant is beyond its predefined design, safety, and operating envelopes; and
- (3) Whether or not conditions that threaten health are expected to be confined to within the site boundary.

The ICs deal explicitly with radiological safety impact by escalating from levels corresponding to releases within regulatory limits to releases beyond EPA Protective Action Guideline (PAG) plume exposure levels. In addition, the "Discussion" sections below include off-site dose consequence considerations which were not included in NUREG-0654 Appendix 1.

**NOTIFICATION OF UNUSUAL EVENT (NOUE):** Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs.

**Discussion:**



Potential degradation of the level of safety of the plant is indicated primarily by exceeding plant technical specification Limiting Condition of Operation (LCO) allowable Action Statement time for achieving required mode change. Precursors of more serious events should also be included because precursors do represent a potential degradation in the level of safety of the plant. Minor releases of radioactive materials are included. In this emergency class, however, releases do not require monitoring or off-site response.

**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.

**Discussion:**

Rather than discussing the distinguishing features of "potential degradation" and "potential substantial degradation," a comparative approach would be to determine whether increased monitoring of plant functions is warranted at the Alert level as a result of safety system degradation. This addresses the operations staff's need for help, independent of whether an actual decrease in plant safety is determined. This increased monitoring can then be used to better determine the actual plant safety state, whether escalation to a higher emergency class is warranted, or whether de-escalation or termination of the emergency class declaration is warranted. Dose consequences from these events are small fractions of the EPA PAG plume exposure levels.

**SITE AREA EMERGENCY (SAE):** 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.

**Discussion:**

The discriminator (threshold) between Site Area Emergency and General Emergency is whether or not the EPA PAG plume exposure levels are expected to be exceeded outside the site boundary. This threshold, in addition to dynamic dose assessment considerations discussed in the EAL guidelines, clearly addresses NRC and off-site emergency response agency concerns as to timely declaration of a General Emergency.

**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 off-site for more than the immediate site area.

**Discussion:**

The bottom line for the General Emergency is whether evacuation or sheltering of the general public is indicated based on EPA PAGs, and therefore should be interpreted to include radionuclide release regardless of cause. In addition, it should address concerns as to uncertainties in systems or

structures (e.g. containment) response, and also events such as waste gas tank releases and severe spent fuel pool events postulated to occur at high population density sites. To better assure timely notification, EALs in this category must primarily be expressed in terms of plant function status, with secondary reliance on dose projection. In terms of fission product barriers, loss of two barriers with loss or potential loss of the third barrier constitutes a General Emergency.

### 3.7 Emergency Class Thresholds

The most common bases for establishing these boundaries are the Technical Specifications, bounding conditions and setpoints for each plant that have been developed in the design basis calculations and the Safety Analysis Report (SAR).

For those conditions that are easily measurable and instrumented, the boundary is likely to be the EAL (observable by plant staff, instrument reading, alarm setpoint, etc.) that indicates entry into a particular emergency class. For example, the main steam line radiation monitor may detect high radiation that triggers an alarm. That radiation level also may be the setpoint that closes the main steam isolation valves (MSIV) and initiates the reactor trip/scram. This same radiation level threshold, depending on plant-specific parameters, also may be the appropriate EAL for a direct entry into an emergency class.

In addition to the continuously measurable indicators, such as coolant temperature, coolant levels, leak rates, containment pressure, etc., the SAR provides indications of the consequences associated with design basis events. Examples would include steam pipe breaks, MSIV malfunctions, and other anticipated events that, upon occurrence, place the plant immediately into an emergency class.

Another approach for defining these boundaries is the use of a plant-specific probabilistic safety assessment (PSA - also known as probabilistic risk analysis, PRA). PRAs have been completed for the designs as part of the licensing process. PRAs can be used as a good first approximation of the relevant ICs and risk associated with emergency conditions. For severe core damage events, uncertainties exist in phenomena important to accident progressions leading to containment failure. Because of these uncertainties, predicting containment integrity may be difficult in these conditions. This is why maintaining containment integrity alone following sequences leading to severe core damage may be an insufficient basis for not escalating to a General Emergency. Establishing EALs for such conditions must take estimated evacuation time into consideration to minimize the potential for the plume to pass while evacuation is underway.

Regardless of whether or not containment integrity is challenged, it is possible for significant radioactive inventory within containment to result in EPA PAG plume exposure levels being exceeded even assuming containment is within technical specification allowable leakage rates. With or without containment challenge, however, a major release of radioactivity requiring off-site protection 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%.

Another critical element of the analysis to arrive at these threshold (boundary) conditions is the time that the plant might stay in that condition before moving to a higher emergency class. The time dimension is critical to the EAL since the purpose of the emergency class for state and local officials is to notify them of the level of mobilization that may be necessary to address the emergency. This is particularly true when a "Site Area Emergency" or "General Emergency" is IMMINENT.

### 3.8 Emergency Action Level Thresholds

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

Classifications are based on evaluation of each Unit. All classifications are to be based upon VALID indications, reports or conditions. Indications, reports or conditions are considered VALID when they are verified by (1) an instrument channel check, or (2) indications on related or redundant indications, or (3) by direct observation by plant personnel, such that doubt related to the indication's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

With the emergency classification levels defined, the thresholds that must be met for each EAL to be placed under the emergency classification level can be determined. There are two basic approaches to determining these EALs. EALs and emergency classification level boundaries coincide for those continuously measurable, instrumented ICs, such as radioactivity, core temperature, coolant levels, etc. For these ICs, the EAL will be the threshold reading that most closely corresponds to the emergency classification level description using the best available information.

For discrete (discontinuous) events, the approach will have to be somewhat different. Typically, in this category are internal and external hazards such as FIRE or earthquake. The purpose for including hazards in EALs is to assure that station personnel and off-site emergency response organizations are prepared to deal with consequential damage these hazards may cause. If, indeed, hazards have caused damage to safety functions or fission product barriers, this should be confirmed by symptoms or by observation of such failures. Therefore, it may be appropriate to enter an Alert status for events approaching or exceeding design basis limits such as earthquake, design basis wind loads, FIRE within VITAL AREAS, etc. This would give the operating staff additional support and improved ability to determine the extent of plant damage. If damage to barriers or challenges to Critical Safety Functions (CSFs) have occurred or are identified, then the additional support can be used to escalate or terminate the emergency classification level based on what has been found. Of course, security events must reflect potential for increasing security threat levels.

Plant emergency operating procedures (EOPs) are designed to maintain and/or restore a set of CSFs which are listed in the order of priority for restoration efforts during accident conditions. The AP1000 set includes:

- Subcriticality
- Core cooling
- Heat sink
- Pressure-temperature-stress (RCS Integrity)
- Containment
- RCS inventory

There are diverse and redundant plant systems to support each CSF. By monitoring the CSFs instead of the individual system component status, the impact of multiple events is inherently addressed, e.g., the number of operable components available to maintain the critical safety function.

The EOPs contain detailed instructions regarding the monitoring of these functions and provides a scheme for classifying the significance of the challenge to the functions. In providing EALs based on these schemes,

the emergency classification level can flow from the EOP assessment rather than being based on a separate EAL assessment. This is desirable as it reduces ambiguity and the time necessary to classify the event.

Emergency Response Guidelines (ERGs) classify challenges as YELLOW, ORANGE, and RED paths. If the core exit thermocouples exceed 1200 degrees F or 700 degrees F with low reactor vessel water level, a RED path condition exists. The ERG considers a RED path as "... an extreme challenge to a plant function necessary for the protection of the public ..." This is almost identical to the present NRC NUREG-0654 description of a site area emergency, "... actual or likely failures of plant functions needed for the protection of the public ..." It reasonably follows that if any CSF enters a RED path, a Site Area Emergency exists. A general emergency could be considered to exist if core cooling CSF is in a RED path and the EOP function restoration procedures have not been successful in restoring core cooling.

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

The ESBWR EOPs contain detailed instructions regarding the monitoring of these functions and provides a scheme for classifying the significance of the challenge to the functions. In providing EALs based on these schemes, the emergency classification can flow from the EOP assessment rather than being based on a separate EAL assessment. This is desirable as it reduces ambiguity and reduces the time necessary to classify the event.

When applicable, a section of the bases designated as developer notes is provided for the development of the site-specific thresholds of the EALs. The information contained in this section consists of references, examples, instructions for calculations, etc. These sections of the basis need not be included in the plant specific technical basis document supporting the EALs. In some cases, the information developed from the developer information may be appropriate to include in the plant specific technical basis document. In addition, the appendices are developer information in their entirety.

### 3.9 Treatment of Multiple Events and Emergency Class Upgrading

The above discussion deals primarily with simpler emergencies and events that may not escalate rapidly. However, usable EAL guidance must also consider rapidly evolving and complex events. Hence, emergency classification level upgrading and consideration of multiple events must be addressed.

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

Emergency classification level upgrading for multi-unit stations must also consider the potential for radioactive release from more than one core at the same site. This must be considered in the emergency classification level declaration.

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

### 3.10 Emergency Classification Level Downgrading

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

### 3.11 Classifying Transient Events

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

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

There may be cases in which a plant condition that exceeded an EAL threshold was not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists. In these cases, an emergency should not be declared.

Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, should be applied.

### 3.12 Operating Mode Applicability

The plant operating mode that existed at the time that the event occurred, prior to any protective system or operator action initiated in response to the condition, is compared to the mode applicability of the EALs. If an event occurs, and a lower or higher plant operating mode is reached before the emergency classification can be made, the declaration shall be based on the mode that existed at the time the event occurred.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that have Cold Shutdown or Refueling for mode applicability, even if Safe/Stable Shutdown (or a higher mode) is entered during any subsequent heatup. In particular, the Fission Product Barrier Matrix EALs are applicable only to events that initiate in Safe/Stable Shutdown or higher.

### MODE APPLICABILITY MATRIX

Mode	Category					
	A	C	E	F	H	S
Operating	X			X	X	X
Startup	X			X	X	X
Hot Standby/ Shutdown	X			X	X	X
Safe/Stable Shutdown	X			X	X	X
Cold Shutdown	X	X			X	
Refueling	X	X			X	
Defueled	X	X			X	
None			X			

#### 3.12.1 ESBWR Operating Modes

Power Operations (1):	Mode Switch in Run.
Startup (2):	Mode Switch in Startup or Refuel.
Hot Shutdown (3):	Mode Switch in Shutdown, Average Reactor Coolant Temperature greater than 420 °F.
Stable Shutdown (4):	Mode Switch in Shutdown, Average Reactor Coolant Temperature less than or equal to 420 °F and greater than 200 °F.
Cold Shutdown (5):	Mode Switch in Shutdown, Average Reactor Coolant Temperature less than or equal to 200 °F.
Refueling (6):	Mode Switch in Shutdown or Refuel, and one or more vessel head bolts less than fully tensioned.
Defueled (None):	All reactor fuel removed from reactor pressure vessel.

#### 3.12.2 AP1000 Operating Modes

Power Operations (1):	Reactor Power greater than 5%, $K_{eff}$ greater than or equal to 0.99.
Startup (2):	Reactor Power less than or equal to 5%, $K_{eff}$ greater than or equal to 0.99.
Hot Standby (3):	RCS greater than 420 °F, $K_{eff}$ less than 0.99.
Safe Shutdown (4):	RCS less than or equal to 420°F, but greater than 200°F , $K_{eff}$ less than 0.99.
Cold Shutdown (5):	RCS less than or equal to 200 °F, $K_{eff}$ less than 0.99.
Refueling (6):	One or more vessel head closure bolts less than fully tensioned.
Defueled (None)	All reactor fuel removed from reactor pressure vessel.

## 4.0 HUMAN FACTORS CONSIDERATIONS

Some factors that should be considered in determining the method of presentation of EALs:

- Who is the audience (user) for this information? A senior utility executive would likely want information presented differently than a licensed operator. Off-site agencies and the NRC may have entirely different information needs.
- The conditions under which the information must be read, understood, and acted upon. Since the subject matter here is *emergency* actions, it is highly likely that the user of the EALs will be under high stress during the conditions where they are required to be used, particularly under conditions corresponding to Site Area Emergency and General Emergency.
- What is the user's perception as to the importance of the EALs compared to other actions and decisions that may be needed at the same time? To allow a licensed operator to discharge his responsibilities for dealing with the situation and also provide prompt notification to outside agencies, the emergency classification and notification process must be rapid and concise.
- Is the EAL consistent with the user's knowledge of what constitutes an *emergency* situation?
- How much help does the user receive in deciding which EAL and emergency class is involved? An Emergency Director with a staffed TSC and EOF has many more resources immediately at his disposal than the licensed operator (typically, the Shift Manager) who has to make the initial decisions and take first actions.

Based on review of a number of plants' EALs and associated information, interviews with utility personnel, and a review of drill experience some recommendations follow.

### 4.1 Level Of Integration Of EALs With Plant Procedures

A rigorous integration of EALs and emergency class determinations into the plant procedure set, although having some benefits, is probably unnecessary. Such a rigorous integration could well make it more difficult to keep documentation up-to-date. However, keeping EALs totally separated from plant procedures and relying on licensed operator or other utility Emergency Director memory during infrequent, high stress periods is insufficient.

RECOMMENDATION: Visual cues in the plant procedures that direct operators to consult the EALs is a method currently used by several utilities. This method can be effective when it is tied to appropriate training. Notes in the appropriate plant procedures to consult the EALs can also be used. It should be noted that this discussion is not restricted to only the emergency procedures; alarm recognition procedures, abnormal operating procedures. Normal operating procedures that apply to cold shutdown and refueling modes should also be included. In addition, EALs can be referenced on entry into particular procedures or existence of particular Critical Safety Function conditions.

### 4.2 Method Of Presentation

A variety of presentation methods are used. Methods range from using procedure language, including specific tag numbers for instrument readings and alarms, deliberate omission of instrument tag numbers, flow charts, critical safety function status trees, checklists, and combinations of the above.

What is clear, however, is that the licensed operator (typically the Shift Manager) is the first user of this information, has the least amount of help in interpreting the EALs, and also has other significant responsibilities to fulfill while dealing with the EALs. Emergency Directors outside the Control Room to

whom responsibilities are turned over have other resources and advisors available to them that a licensed operator may not have when first faced with an emergency situation. In addition, as an emergency situation evolves, the operating staff and the health physics staff are the personnel who must first deal with information that is germane to changing the emergency classification (up, down, or out of the emergency class).

**RECOMMENDATION:** The method of presentation should be one with which the operations and health physics staff are comfortable. As is the case for emergency procedures, bases for steps should be in a separate (or separable) document suitable for training and for reference by emergency response personnel and off-site agencies. Each nuclear plant should already have presentation and human factors standards as part of its procedure writing guidance. EALs that are consistent with those procedure writing standards (in particular, emergency operating procedures which most closely correspond to the conditions under which EALs must be used) should be the norm for each utility.

The presentation method often used in the industry is one in which the ICs, applicable operating modes, and example EALs are laid out on a large wall board and grouped by category, or sub-category. Usually two EAL boards are developed such that one board will have all applicable EALs for HOT operating modes and one board will have all applicable EALs for COLD operating modes. An EAL Basis Document is also developed. The EAL Basis Document will contain the same information and will also have relevant information included to support an understanding of the EAL and EAL methodology.

#### 4.3 Symptom-based, Event-based, Or Barrier-based EALs

A review of the emergency class descriptions provided elsewhere in this document shows that NOUEs and Alerts deal primarily with sequences that are precursors to more serious emergencies or that may have taken a plant outside of its intended operating envelope, but currently pose no danger to the public. Observable indications in these classes 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 one escalates to Site Area Emergency and General Emergency, potential radiological impact to people (both on-site and off-site) rises. However, at this point the root cause event(s) leading to the emergency class escalation matter far less than the increased (potential for) radiological releases. Thus, EALs for these emergency classes should be primarily symptom- and barrier-based. It should be noted again, as stated in Section 3.3, that barrier monitoring is a subset of symptom monitoring, i.e., what readings (symptoms) indicate a challenge to a fission product barrier.

**RECOMMENDATION:** A combination approach that ranges from primarily event-based for NOUEs to primarily symptom- or barrier-based for General Emergencies is recommended. This is to better assure that timely recognition and notification occurs, that events occurring during refueling and cold shutdown are appropriately covered, and that multiple events can be effectively treated in the EALs.



## 5.0 GENERIC EAL GUIDANCE

This section provides generic EAL guidance based on the information gathered and reviewed by the Task Force. Because of the wide variety of presentation methods used at different utilities, this document specifies guidance as to what each IC and EAL should address, and including sufficient basis information for each will best assure uniformity of approach. This approach is analogous to reactor vendors' owners groups developing generic emergency procedure guidelines which are converted by each utility into plant-specific emergency operating procedures. Each utility is reminded, however, to review the "Human Factors Considerations" section of this document as part of implementation of the attached Generic EAL Guidance.

### 5.1 Generic Arrangement

The information is presented by Recognition Categories:

- A - Abnormal Rad Levels / Radiological Effluent
- C - Cold Shutdown / Refueling System Malfunction
- D - Permanently Defueled Station Malfunction
- E - Events Related to Independent Spent Fuel Storage Installations
- F - Fission Product Barrier Degradation
- H - Hazards or Other Conditions Affecting Plant Safety
- S - System Malfunction

The Initiating Conditions for each of the above Recognition Categories are in the order of NOUE, Alert, Site Area Emergency, and General Emergency. For all Recognition Categories, an Initiating Condition matrix versus Emergency Class is first shown. For Recognition Category F, the barrier-based EALs are presented in Tables 5-F-2 and 5-F-3 for ESBWR and AP1000 respectively.

With the exception of Recognition Category F, each of the EAL guides in Recognition Categories is structured in the following way:

- **Recognition Category** - As described above.
- **Emergency Class** - NOUE, Alert, Site Area Emergency or General Emergency.
- **Initiating Condition** – Symptom- or Event-Based, Generic Identification and Title.
- **Operating Mode Applicability** - These modes are defined in each licensee's Technical Specifications. The mode classifications and terminology appropriate to the specific facility should be used.
- **Example Emergency Action Level(s)** – These EALs are examples of conditions and indications that were considered to meet the criteria of the IC. For site specific implementation, the examples are Threshold Values that become an Emergency Action Level that can be classified based on the relative threat the condition represents.
- **Basis** – Provides information that explains the IC and example EALs. The bases are written to assist the personnel implementing the generic guidance into site-specific procedures. Some bases provide information intended to assist with establishing site-specific instrumentation values. Where needed, appropriate developer information is provided as a section of the basis.

For Recognition Category F, basis information is presented in a format consistent with Tables 5-F-1, 2 and 3. The presentation method shown for Fission Product Barrier Function Matrix was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. Other acceptable methods of achieving these goals which are currently in use include flow charts, block diagrams, and checklist tables. Licensees selecting these alternatives need to ensure that all possible EAL combinations in the Fission Product Barrier Function Table are addressed in their presentation method.

### 5.2 Generic Bases

The generic guidance has the primary threshold for NOUEs as operation outside the safety envelope for the plant as defined by plant Technical Specifications, including LCOs and Action Statement Times. In addition, certain precursors of more serious events are included in NOUE IC/EALs. This provides a clear demarcation between the lowest emergency class and "non-emergency" notifications specified by 10 CFR 50.72.

For a number of Alerts, IC/EALs are chosen based on hazards which may cause damage to plant safety functions (i.e., tornadoes, hurricanes, FIRE in plant VITAL AREAs) or require additional help directly (Control Room evacuation) and thus increased monitoring of the plant is warranted. The symptom-based and barrier-based IC/EALs are sufficiently anticipatory to address the results of multiple failures, regardless of whether there is or is not a common cause. Declaration of the Alert will already result in the staffing of the TSC for assistance and additional monitoring. Thus, direct escalation to the Site Area Emergency is unnecessary. Other Alerts, that have been specified, correspond to conditions which are consistent with the emergency class description.

The basis for declaring a Site Area Emergency and General Emergency is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.

With regard to the Hazards Recognition Category, the existence of a hazard that represents a potential degradation in the level of safety of the plant is the basis of NOUE classification. If the hazard results in VISIBLE DAMAGE to plant structures or equipment associated with safety systems or if system performance is affected, the event may be escalated to an Alert. The reference to "duration" or to "damage" to safety systems is intended only to size the event. Consequential damage from such hazards, if observed, would be the basis for escalation to Site Area Emergency or General Emergency, by entry to System Malfunction or Fission Product Barrier IC/EALs.

### 5.3 Site-Specific Implementation

The generic guidance includes both ICs and example EALs. It is the intent of this guidance that both be included in the site-specific implementation. Each serves a specific purpose. The IC is intended to be the fundamental criteria for the declaration, whereas, the EALs are intended to represent unambiguous examples of conditions, Threshold Values, that may meet the IC. There may be unforeseen events, or combinations of events, for which the EALs may not be exceeded, but in the judgment of the Emergency Director, the intent of the IC may be met. While the generic guidance does include Emergency Director judgment ICs, the additional detail in the individual ICs will facilitate classifications over the broad guidance of the Emergency Director judgment ICs.

Even though passive reactor design does not incorporate shared safety systems or functions, sites involving more than one reactor unit, should consider how events involving one unit (such as radiological release) may affect other units, and whether or not this may be a factor in escalating the event.

State and local requirements have not been reflected in the generic guidance and should be considered on a case-by-case basis with appropriate state and local emergency response organizations.

Utilities should prepare a basis document or include basis information with the IC/EALs. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information may be useful in training, for explaining event classifications to off-site officials, and would facilitate regulatory review and approval of the classification scheme.

### 5.4 Definitions

In the IC/EALs, selected words have been set in all capital letters. These words are defined terms having specific meanings as they relate to this document. Definitions of these terms are provided below. Definitions

preceded by an asterisk (\*) are not used in this document but are included to provide common terminology though out the nuclear industry.

**\*AFFECTING SAFE SHUTDOWN:** Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable HOT or COLD SHUTDOWN condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in HOT SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is not “AFFECTING SAFE SHUTDOWN.”

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in COLD SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is “AFFECTING SAFE SHUTDOWN.”

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

**\*CIVIL DISTURBANCE:** A group of persons violently protesting station operations or activities at the site.

**CONFINEMENT BOUNDARY:** The barrier(s) between areas containing radioactive substances and the environment.

**CONTAINMENT CLOSURE:** (AP1000) The Tech. Spec. Sections 3.6 and 3.9 contain the required and site specific procedurally defined actions necessary to secure primary containment and the associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. (ESBWR) The Tech. Spec. Section 3.6 required and site-specific procedurally defined action taken to secure primary containment or the Reactor Building and the associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions.

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

**\*EXTORTION:** An attempt to cause an action at the station by threat of force.

**FAULTED:** (AP1000) in a steam generator, the existence of secondary side leakage that results in an uncontrolled drop in steam generator pressure or the steam generator being completely depressurized.

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

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

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

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

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

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

**ISFSI (Independent Spent Fuel Storage Installation):** 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 PLANT OPERATIONS:** Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

**OWNER CONTROLLED AREA:** [Site-specific] Typically, all the property owned and controlled by the company outside the protected area, security fence line that immediately surrounds the plant. Access is normally limited to persons entering for official business.

**PROJECTILE:** An object directed toward a NPP that could have an effect sufficient to cause concern for its continued operability, reliability, or safety of personnel.

**PROTECTED AREA:** [Site-specific] Typically, the area which normally encompasses all controlled areas within the security PROTECTED AREA fence.

**REACTOR BUILDING ISOLATION:** (ESBWR) See CONTAINMENT CLOSURE.

**RUPTURED:** (AP1000) in a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and automatic depressurization.

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

**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.

**\*STRIKE ACTION:** A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on (site specific). The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONS.

**UNISOLABLE:** A breach or leak that cannot be promptly isolated.

**UNPLANNED:** A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

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

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

**VITAL AREA:** [Site-specific] Typically, any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

5.5 ABNORMAL RAD LEVELS/ RADIOLOGICAL EFFLUENT EALs

**Table 5-A-1: Recognition Category “A” Initiating Condition Matrix**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
<p><b>AG1</b> Off-site Dose Resulting from an Actual or IMMEDIATE Release of Gaseous Radioactivity Greater Than 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. <i>Op. Modes: All</i></p>	<p><b>AS1</b> Off-site Dose Resulting from an Actual or IMMEDIATE Release of Gaseous Radioactivity Greater Than 100 mrem TEDE or 500 mrem Thyroid CDE for the Actual or Projected Duration of the Release. <i>Op. Modes: All</i></p>	<p><b>AA1</b> Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 200 Times the Off-site Dose Calculation Manual for 15 Minutes or Longer. <i>Op. Modes: All</i></p>	<p><b>AU1</b> Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 2 Times the Off-site Dose Calculation Manual for 60 Minutes or Longer. <i>Op. Modes: All</i></p>
		<p><b>AA3</b> Rise In Radiation Levels Within the Facility that Impedes Operation Of Systems Required to Maintain Plant Safety Functions. <i>Op. Modes: All</i></p>	<p><b>AU2</b> UNPLANNED Rise in Plant Radiation Levels. <i>Op. Modes: All</i></p>
		<p><b>AA2</b> 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. <i>Op. Modes: All</i></p>	

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 2 Times the Off-site Dose Calculation Manual for 60 Minutes or Longer.

**Operating Mode Applicability:** **All**

**Example Emergency Action Level Thresholds: (1 or 2 or 3 or 4 or 5)**

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

1. VALID reading on any of the following radiation monitors indicating greater than 2 times the ODCM setpoint for 60 minutes or longer:

AP1000

Steam Generator Blowdown	BDS-RE-010	[Site specific]
	BDS-RE-011	[Site specific]
Main Steam Line	SGS-RIA-026, RIA-027	[Site specific]
Service Water Blowdown	SWS-RIA-008	[Site specific]
Containment Air Filtration Exhaust	VFS-MA-02A, MA-02B	[Site specific]

ESBWR

Main Steamline	D11-PRM-RMS-01	[Site specific]
Containment Purge Exhaust	D11-PRM-RMS-23	[Site specific]
Drywell Sump LCW/HCW Discharge	D11-PRM-RMS-16	[Site specific]
Turbine Bldg. Combined Ventilation Exhaust	D11-PRM-RMS-10	[Site specific]
Radwaste Bldg. Ventilation Exhaust	D11-PRM-RMS-17	[Site specific]

2. VALID reading on any of the following radiation monitors indicating greater than 2 times the ODCM setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

AP1000

Plant Vent	VFS-RICA-103	[Site specific]
Turbine Island Vent	TDS-JE-RE001	[Site specific]
Gaseous Radwaste Discharge	WGS-RICA-017	[Site specific]
Liquid Radwaste discharge	WLS-RIA-229	[Site specific]
Wastewater Discharge	WWS-JE-RE021	[Site specific]

ESBWR

Reactor Bldg/Fuel Bldg Stack	D11-PRM-RMS-13A	[Site specific]
Turbine Building Stack	D11-PRM-RMS-13B	[Site specific]
Rad Waste Building Stack	D11-PRM-RMS-13C	[Site specific]
Liquid Radwaste Discharge	D11-PRM-RMS-11	[Site specific]
Isolation Condenser Vent Exhaust	D11-PRM-RMS-19	[Site specific]

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 2 times [Site specific ODCM values] for 60 minutes or longer.

## ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

4. VALID reading on perimeter radiation monitoring system greater than 0.10 mR/hr above normal\* background for 60 minutes or longer.

\*Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

5. VALID indication on automatic real-time dose assessment capability indicating greater than 0.10 mr/hr at or beyond the site boundary for 60 minutes or longer.

### **Basis:**

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

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

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Off-site Dose Calculation Manual (ODCM)<sup>1</sup>. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

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

Releases should not be prorated or averaged. For example, a release exceeding 4 times ODCM for 30 minutes does not meet the threshold for this IC.

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

Threshold #1 addresses effluent or accident radiation monitors on non-routine release pathways (i.e., for which a discharge permit would not normally be prepared). The setpoint is based on radiation monitor exceeding two times the ODCM limit and releases are not terminated within 60 minutes. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments.

Threshold #2 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the ODCM limit and releases are not terminated within 60 minutes. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release. Indexing the EAL threshold to the ODCM setpoints in this manner ensures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

Threshold #3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.

EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints. EALs #4 and #5 are a

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<sup>1</sup> For the AP1000 and the ESBWR the ODCM will conform to NEI 07-09A



## ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency.

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

### **Developer Notes:**

Thresholds #1 and # 2 – Calculate the site specific ODCM setpoint using guidance provided in Off-site Dose Calculation Manual sections 6.2.2 and 6.3.2. Fill in the site specific setpoint that is two times the ODCM setpoint calculated.

Threshold #3 – Calculate the site specific ODCM setpoint using guidance provided in Off-site Dose Calculation Manual sections 6.2.1 and 6.3.1. Fill in the site specific value that is two times the ODCM limit.

Threshold #4 – If the site does not have a perimeter monitoring system then delete the threshold from the IC. If the site does have a perimeter monitoring system then use Threshold #4 as shown and add the following to the basis following the Threshold 3 description paragraph. “Threshold #4 value of 0.10 mR/hr is based on a release rate not exceeding 500 mrem per year. As provided in the ODCM , prorated over 8766 hours, multiplied by two, and rounded. ( $500 \div 8766 \times 2 = 0.114$ ). The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #4 is an indication of an uncontrolled release.”

Threshold #5 – If the site does not have real-time dose assessment capability then delete the threshold from the IC. If the site does have real-time dose assessment capability then use Threshold #5 as shown and add the following to the basis following the Threshold #4 description paragraph. “Threshold #4 value of 0.10 mR/hr is based on a release rate not exceeding 500 mrem per year. As provided in the ODCM , prorated over 8766 hours, multiplied by two, and rounded. ( $500 \div 8766 \times 2 = 0.114$ ). The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #5 is an indication of an uncontrolled release.”

### AP1000 References:

VFS-M3C-101  
WGS-M3C-101  
WLS-M3C-101  
WWS-M3C-100  
BDS-M3C-101  
SGS-M3C-101  
SWS-M3C-101  
RMS-J7-001

### ESBWR References:

DCD Tier 2, Section 11.5.1  
DCD Tier 2, Table 11.5-3  
DCD Tier 2, Figure 11.5-1

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

UNPLANNED Rise in Plant Radiation Levels.

**Operating Mode Applicability: All**

**Example Emergency Action Level Thresholds: (1 or 2)**

- 1. a. UNPLANNED water level drop in a refueling pathway as indicated by any of the following:

AP1000

- Spent Fuel Pool Low Alarm on SFS-LICA-19A/B/C
- Visual observation

ESBWR

Rx Well Cavity	G21-FAPCS-LS-N020-Low Alarm
Buffer Pool	G21-FAPCS-LS-N019-Low Alarm
Upper Fuel Transfer Pool	G21-FAPCS-LS-N018-Low Alarm
Skimmer Surge Tank A/B Level	G21-FAPCS-LS-R621-Low Alarm
	G21-LS-R622-Low Alarm
Spent Fuel Storage Pool	G21-FAPCS-LS-R634-Low Alarm
	G21-LS-R632, G21-LS-R633-Low Alarm
Lower Fuel Transfer Pool	G21-FAPCS-LS-N026-Low Alarm
Visual observation	

**AND**

- b. VALID Area Radiation Monitor reading rise on any of the following:

AP1000

Fuel Handling Area Exhaust Radiation Monitor	VAS-RE 001
Containment High Range	PXS-RICA-160, 161, 162, 163
Refueling Bridge Portable Monitor (Refueling Mode Only)	[Site specific]

ESBWR

Refueling Floor Area #1, EL 34000 (Reactor Building)	[Site specific]
Refueling Floor Area #2, EL 34000 (Reactor Building)	[Site specific]
New Fuel Buffer Pool, EL 27000 (Reactor Building)	[Site specific]
New Fuel Buffer Pool, EL 27000 (Reactor Building)	[Site specific]
Fuel Handling Machine (IFTS), EL 34000 (Reactor Building)	[Site specific]
Spent Fuel Floor, EL 4650 (Fuel Building)	[Site specific]
Fuel Handling Machine, EL 4650, (Fuel Building)	[Site specific]
Fuel Transfer Cask Area, EL 4650 (Fuel Building)	[Site specific]
IFTS Fuel Building Isolation Valve Room (Inside)	[Site specific]

- 2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal\* levels.

\*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

## ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

### **Basis:**

This IC addresses elevated radiation levels as a result of water level lowering but above the RPV flange or events that have resulted, or may result, in UNPLANNED rise in radiation dose rates within plant buildings. These radiation levels represent a loss of control over radioactive material and may represent a potential degradation in the level of safety of the plant.

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

The refueling pathway is a site specific combination of cavities, tubes, canals and pools. While a radiation monitor could detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the refueling bridge portable monitor is a portable area monitor secured to the refueling bridge only during refueling operations and is therefore only applicable in the Refueling Mode. The reading on this refueling bridge area radiation monitor may rise due to planned evolutions such as removal of the reactor head, a fuel assembly being raised in the manipulator mast, or could in fact be properly responding to a known event involving transfer or relocation of a source stored in or near the fuel pool. Generally, elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss. Application of this EAL requires understanding of the actual radiological conditions present in the vicinity of the monitor.

For refueling events where the water level drops below the RPV flange classification would be via CU2. This event escalates to an Alert per IC 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 Matrix for events in operating modes 1-4.

Threshold #1 addresses unexpected loss of refueling pathway level that could be a potential precursor to increased radiation levels and which, without operator action, could potentially challenge one of the safety barriers with respect to the refueling pathway.

The ESBWR fuel pool cooling function is also provided in the event that a recently unloaded fuel batch requires continued cooling during the post-accident period. The spent fuel pool contains sufficient inventory to ensure no operator action is required during the first 72 hours. After that period, either makeup water must be supplied to the spent fuel pool or the FAPCS must be initiated. The FAPCS equipment is environmentally qualified, so access is not required and redundancy is included in system components.

If available, video cameras may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or decrease in refueling water storage tank level.

Threshold #2 addresses elevated in-plant radiation levels encountered during operation of plant processes that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. This EAL excludes in-plant radiation levels that may result from use of radiographic sources. A specific list of ARMs is not required as it would restrict the applicability of the Threshold. The intent is to identify loss of control of radioactive material in any monitored area.

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

**Developer Notes:**

AP1000 Threshold #1b – The refueling bridge portable monitor will normally be secured to the refueling bridge only during refueling operations. Since it is not a permanent plant monitor, this monitor is not within the scope of supply for the RMS for the AP1000 design. Fill in the instrument type to be used when a determination of instrument type to be used is established.

ESBWR Threshold #1b – Fill in the instrument number when it has been established.

AP1000 References:

SFS-M3C-101  
RCS-M3C-101  
VAS-M3C-101  
PXS-M3C-101  
RMS-J7-001

ESBWR References:

DCD Tier 1, Section 2.6.2  
DCD Tier 2, Section 9.1.3  
DCD Tier 2, Figure 9.1-1  
DCD Tier 2, Table 12.3-2  
DCD Tier 2, Table 12.3-3  
TS 3.7.5  
TS 3.9.6

**Initiating Condition -- ALERT**

Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 200 Times the Off-site Dose Calculation Manual for 15 Minutes or Longer.

**Operating Mode Applicability: All**

**Example Emergency Action Level Thresholds: (1 or 2 or 3 or 4 or 5)**

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

1. VALID reading on any of the following radiation monitors indicating greater than 200 times the ODCM setpoint for 15 minutes or longer :

AP1000

Steam Generator Blowdown	BDS-RE-011	[Site specific]
	BDS-RE-010	[Site specific]
Main Steam Line	SGS-RIA-026, RIA-027	[Site specific]
Service Water Blowdown	SWS-RIA-008	[Site specific]
Containment Air Filtration Exhaust	VFS-MA-02A, MA-02B	[Site specific]

ESBWR

Main Steamline	D11-PRM-RMS-01	[Site specific]
Containment Purge Exhaust	D11-PRM-RMS-23	[Site specific]
Drywell Sump LCW/HCW Discharge	D11-PRM-RMS-16	[Site specific]
Turbine Bldg. Combined Ventilation Exhaust	D11-PRM-RMS-10	[Site specific]
Radwaste Bldg. Ventilation Exhaust	D11-PRM-RMS-17	[Site specific]

2. VALID reading on any of the following radiation monitors indicating greater than 200 times the ODCM setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

AP1000

Plant Vent	VFS-RICA-103	[Site specific]
Turbine Island Vent	TDS-JE-RE001	[Site specific]
Gaseous Radwaste Discharge	WGS-RICA-017	[Site specific]
Liquid Radwaste discharge	WLS-RIA-229	[Site specific]
Wastewater Discharge	WWS-JE-RE021	[Site specific]

ESBWR

Reactor Bldg/ Fuel Bldg Stack	D11-PRM-RMS-13A	[Site specific]
Turbine Building Stack	D11-PRM-RMS-13B	[Site specific]
Rad Waste Building Stack	D11-PRM-RMS-13C	[Site specific]
Liquid Radwaste Discharge	D11-PRM-RMS-11	[Site specific]
Isolation Condenser Vent Exhaust	D11-PRM-RMS-19	[Site specific]

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates, greater than 200 times [site specific ODCM values] for 15 minutes or longer.

## ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

4. VALID reading on perimeter radiation monitoring system greater than 10.0 mR/hr above normal\* background for 15 minutes or longer.  
\*Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.
5. VALID indication on automatic real-time dose assessment capability indicating greater than 10.0 mr/hr at or beyond the site boundary for 15 minutes or longer.

### **Basis:**

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

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

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Off-site Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in the features and/or controls established to prevent unintentional releases, or control and monitor intentional releases.

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

Releases should not be prorated or averaged.

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

Threshold #1 addresses effluent or accident radiation monitors on non-routine release pathways (i.e., for which a discharge permit would not normally be prepared).

Threshold #2 addresses radioactivity releases that for whatever reason cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. This alarm setpoint may be associated with a planned batch release, or a continuous release path.

In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance. Indexing the EAL threshold to the ODCM setpoints in this manner ensures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints.

Threshold #3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.

**Developer Notes:**

Thresholds #1 and #2 – Calculate the site specific ODCM setpoint using guidance provided in Off-site Dose Calculation Manual sections 6.2.2 & 6.3.2. Fill in the site specific setpoint that is two hundred times the ODCM setpoint calculated..

The value will be based on radiation monitor readings that exceed 200 times the ODCM limit. The developer should ensure a realistic near-linear escalation alarm value has been developed roughly half-way between the AU1 value and the value calculated for AS1 to ensure that the Alert does not exceed the Site Area Emergency threshold. If the value determined for 200 times ODCM is not roughly half-way between the AU1 and AS1 values, a value should be provided that represents a near linear half value. The IC and the basis sections must be modified to describe the revised threshold.

Threshold #3 – Calculate the site specific ODCM setpoint using guidance provided in Off-site Dose Calculation Manual sections 6.2.1 and 6.3.1. Fill in the site specific value that is 200 times the ODCM limit.

Threshold #4 – If the site does not have a perimeter monitoring system then delete the threshold from the IC. If the site does have a perimeter monitoring system then use Threshold #4 as shown and add the following to the basis following the Threshold 3 description paragraph. “Threshold #4 value of 10.0 mR/hr is based on a release rate not exceeding 500 mrem per year. As provided in the ODCM, prorated over 8766 hours, multiplied by 200, and rounded. ( $500 \div 8766 \times 200 = 11.4$ ). The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #4 is an indication of an uncontrolled release. EAL #4 is a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency.”

Threshold #5 – If the site does not have real-time dose assessment capability then delete the threshold from the IC. If the site does have real-time dose assessment capability then use Threshold #5 as shown and add the following to the basis following the Threshold #4 description paragraph. “Threshold #5 value of 10.0 mR/hr is based on a release rate not exceeding 500 mrem per year. As provided in the ODCM, prorated over 8766 hours, multiplied by 200, and rounded. ( $500 \div 8766 \times 200 = 11.4$ ). The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #5 is an indication of an uncontrolled release. EAL #5 is a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency.”

AP1000 References:

VFS-M3C-101  
WGS-M3C-101  
WLS-M3C-101  
WWS-M3C-100  
BDS-M3C-101  
SGS-M3C-101  
SWS-M3C-101  
RMS-J7-001

ESBWR References:

DCD Tier 2, Section 11.5.1  
DCD Tier 2, Table 11.5-3  
DCD Tier 2, Figure 11.5-1

**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 Level Thresholds: (1 or 2)**

1. A water level drop in the reactor refueling pathway resulting in irradiated fuel becoming uncovered as indicated by any of the following:

AP1000

Spent Fuel Pool Low-Low Alarm	SFS-LICA-19A/B/C
Visual observation	

ESBWR

Rx Well Cavity	G21-FAPCS-LS-N020-Low Low Alarm
Buffer Pool	G21-FAPCS-LS-N019-Low Low Alarm
Upper Fuel Transfer Pool	G21-FAPCS-LS-N018-Low Low Alarm
Skimmer Surge Tank A/B Level	G21-FAPCS-LS-R621-Low Low Alarm
	G21-FAPCS-LS-R622-Low Low Alarm
Spent Fuel Storage Pool	G21-FAPCS-LS-R634-Low Low Alarm
	G21-FAPCS-LS-R632-Low Low Alarm
	G21-FAPCS-LS-R633-Low Low Alarm
Lower Fuel Transfer Pool	G21-FAPCS-LS-N026-Low Low Alarm
Visual observation	

2. A VALID alarm or elevated reading on any of the following due to damage to irradiated fuel or loss of water level:

AP1000

Fuel Handling Area Exhaust Radiation Monitor	VAS-RE-001
Containment High Range	PXS-RICA-160,161,162, 163
Refueling Bridge Portable Monitor (Refueling Mode Only)	[Site specific]

ESBWR

Refueling Floor Area #1, EL 34000 (Reactor Building)	[Site specific]
Refueling Floor Area #2, EL34000 (Reactor Building)	[Site specific]
New Fuel Buffer Pool, EL 27000 (Reactor Building)	[Site specific]
New Fuel Buffer Pool, EL 27000 (Reactor Building)	[Site specific]
Fuel Handling Machine (IFTS), EL 34000 (Reactor Building)	[Site specific]
Spent Fuel Floor, EL 4650 (Fuel Building)	[Site specific]
Fuel Handling Machine, EL 4650 (Fuel Building)	[Site specific]
Fuel Transfer Cask Area, EL 4650 (Fuel Building)	[Site specific]
IFTS Fuel Building Isolation Valve Room (Inside), EL 4650	[Site specific]



## ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

### **Basis:**

This IC addresses specific events that have resulted, or may result, in unexpected rise in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent degradation in the level of safety of the plant.

These events escalate from IC AU2 in that fuel activity has been released, or is anticipated due to fuel heatup.

In Threshold #1, site-specific indications includes instrumentation for water level and personnel (e.g., refueling crew) reports. The refueling pathway is a site specific combination of cavities, tubes, canals and pools. If available, video cameras may allow remote observation. Depending on available level indication, the declaration threshold may need to be based on indications of water makeup rate.

Threshold #2 addresses radiation monitor indications of fuel uncover and/or fuel damage.

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

The refueling pathway is a site specific combination of cavities, tubes, canals and pools. While a radiation monitor could detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the refueling bridge portable monitor is a portable area monitor secured to the refueling bridge only during refueling operations and is therefore only applicable in the Refueling Mode. The reading on this refueling bridge area radiation monitor may rise due to planned evolutions such as removal of the reactor head, a fuel assembly being raised in the manipulator mast, or could in fact be properly responding to a known event involving transfer or relocation of a source stored in or near the fuel pool. Generally, elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss. Application of this EAL requires understanding of the actual radiological conditions present in the vicinity of the monitor.

Escalation, if appropriate, would occur via IC AS1 or AG1.

### **Developer Notes:**

AP1000 Threshold #2 – The refueling bridge portable monitor will normally be secured to the refueling bridge only during refueling operations. Since it is not a permanent plant monitor, this monitor is not within the scope of supply for the RMS for the AP1000 design. Fill in the instrument type to be used when a determination of instrument type to be used is established.

ESBWR Threshold #2 - Fill in the instrument number when it has been established.

#### AP1000 References:

SFS-M3C-101  
VAS-M3C-101  
PXS-M3C-101  
RMS-J7-001

#### ESBWR References:

DCD Tier 1, Section 2.6.2  
DCD Tier 2, Section 9.1.3  
DCD Tier 2, Figure 9.1-1  
DCD Tier 2, Table 12.3-2  
DCD Tier 2, Table 12.3-3  
TS 3.7.5  
TS 3.9.6

**Initiating Condition -- ALERT**

Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Plant Safety Functions.

**Operating Mode Applicability:** All

**Example Emergency Action Level Thresholds:**

- 1. Dose rate greater than 15 mR/hr in the following areas requiring continuous occupancy to maintain plant safety functions:

AP1000

Main Control Room	RMS-JE-RE010
Central Alarm Station	RMS-JE-RE009

ESBWR

Main Control Room	[Site specific]
Central Alarm Station	[Site specific]

**Basis:**

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

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

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

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

The value of 15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section II.B.2 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.

For the AP1000, the Containment Area–Personnel Hatch radiation monitor, RMS-JE-RE009, monitors the area in immediate proximity to Rooms 12451, 12452 and 12454. This monitor would be used to alert Security personnel associated with the Central Alarm Station (CAS).

**Developer Notes:**

ESBWR

Threshold #1 – Fill in the appropriate radiation instrument numbers when they become established.

AP1000 References:  
RMS-J7-001

ESBWR References:  
DCD Tier 2, Table 12.3-6  
DCD Tier 2, Figure 12.3-25

**Initiating Condition -- SITE AREA EMERGENCY**

Off-site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 100 mrem TEDE or 500 mrem Thyroid CDE for the Actual or Projected Duration of the Release.

**Operating Mode Applicability:** All

**Example Emergency Action Level Thresholds: (1 or 2 or 3 or 4)**

*Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on any of the following radiation monitors greater than the reading shown for 15 minutes or longer:

AP1000

Plant Vent (Mid Range Gas)	VFS-RIA-104A	[Site specific]
Plant Vent (High Range Gas)	VFS-RIA-104B	[Site specific]
Gaseous Radwaste Discharge	WGS-RICA-017	[Site specific]

ESBWR

Reactor Bldg/ Fuel Bldg Stack	D11-PRM-RMS-13A	[Site specific]
Turbine Building Stack	D11-PRM-RMS-13B	[Site specific]
Rad Waste Building Stack	D11-PRM-RMS-13C	[Site specific]
Isolation Condenser Vent Exhaust	D11-PRM-RMS-19	[Site specific]

2. Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary.
3. A VALID perimeter radiation monitoring system reading greater than 100 mR/hr for 15 minutes or longer.
4. Field survey results indicate closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation, at or beyond the site boundary.

**Basis:**

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

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

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

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

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

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

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

**Developer Notes:**

For Threshold #1, the monitor reading should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site specific boundary (or beyond) dose of 100 mrem whole body or 500 mrem thyroid in one hour, whichever is more limiting. If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.

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

Threshold #3 – Delete Threshold #3 if the site does not have a perimeter monitoring system. If the site does have a perimeter monitoring system then add the following to the end of the basis. "The 100.0 mR/hr value in Threshold #3 is based on a release rate not exceeding 500 mrem per year. As provided in the ODCM, prorated over 8766 hours, multiplied by 2000, and rounded. ( $500 \div 8766 \times 2000 = 114$ )."

AP1000 References:

VFS-M3C-101  
WGS-M3C-101  
RMS-J7-001

ESBWR References:

DCD Tier 2, Section 11.5.1  
DCD Tier 2, Table 11.5-3  
DCD Tier 2, Figure 11.5-1

**Initiating Condition -- GENERAL EMERGENCY**

Off-site Dose Resulting from an Actual or IMMEDIATE Release of Gaseous Radioactivity Greater Than 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

**Operating Mode Applicability:** All

**Example Emergency Action Level Thresholds: (1 or 2 or 3 or 4)**

*Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on any of the following radiation monitors greater than the reading shown for 15 minutes or longer:

AP1000

Plant Vent (Mid Range Gas)	VFS-RIA-104A	[Site specific]
Plant Vent (High Range Gas)	VFS-RIA-104B	[Site specific]
Gaseous Radwaste Discharge	WGS-RICA-017	[Site specific]

ESBWR

Reactor Bldg/ Fuel Bldg Stack	D11-PRM-RMS-13A	[Site specific]
Turbine Building Stack	D11-PRM-RMS-13B	[Site specific]
Rad Waste Building Stack	D11-PRM-RMS-13C	[Site specific]
Isolation Condenser Vent Exhaust	D11-PRM-RMS-19	[Site specific]

2. Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary.
3. VALID perimeter radiation monitoring system reading greater than 1000 mR/hr. for 15 minutes or longer.
4. Field survey results indicate closed window dose rates greater than 1000 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for one hour of inhalation, at or beyond site boundary.

**Basis:**

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

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

While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the

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more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

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

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

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

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

### **Developer Notes:**

The EPA PAG guidance provides for the use of adult thyroid dose conversion factors. However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facilities emergency planning zone.

The monitor reading EALs should be determined using a dose assessment method that back-calculates from the dose values specified in the IC. The meteorology and source term (noble gases, particulates, and halogens) used should be the same as those used for determining the monitor reading EALs in ICs AU1 and AA1. This protocol will maintain intervals between the EALs for the four classifications. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site specific boundary (or beyond) dose of 1000 mrem whole body or 5000 mrem thyroid, whichever is more limiting (as was done for EALs #3 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.

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

Threshold #3 – Delete Threshold #3 if the site does not have a perimeter monitoring system. If the site does have a perimeter monitoring system then add the following to the end of the basis. "The 1000.0 mR/hr value in Threshold #3 is based on a release rate not exceeding 500 mrem per year. As provided in the ODCM, prorated over 8766 hours, multiplied by 20000, and rounded.  $(500 \div 8766 \times 20000 = 1140)$ ."

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AP1000 References:

VFS-M3C-101  
WGS-M3C-101  
RMS-J7-001

ESBWR References:

DCD Tier 2, Section 11.5.1  
DCD Tier 2, Table 11.5-3  
DCD Tier 2, Figure 11.5-1

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

5.6 COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION EALS

**Table 5-C-1: Recognition Category “C” Initiating Condition Matrix**

<b>GENERAL EMERGENCY</b>	<b>SITE AREA EMERGENCY</b>	<b>ALERT</b>	<b>NOUE</b>
<b>CG1</b> Loss of RCS/RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged. <i>Op. Modes: Cold Shutdown, Refueling</i>	<b>CS1</b> Loss of RCS/RPV Inventory Affecting Core Decay Heat Removal Capability. <i>Op. Modes: Cold Shutdown, Refueling</i>	<b>CA1</b> Loss of RCS/RPV Inventory. <i>Op. Modes: Cold Shutdown, Refueling</i>	<b>CU1</b> RCS Leakage. <i>Op. Mode: Cold Shutdown</i>
		<b>CA2</b> Loss of all Off-site and all On-site AC power to Busses ECS-ES-1 and 2 (AP1000) PIP Busses (ESBWR) for 60 minutes or longer. <i>Op. Modes: Cold Shutdown, Refueling, Defueled</i>	<b>CU2</b> UNPLANNED Loss of RCS/RPV Inventory. <i>Op. Mode: Refueling</i>
		<b>CA3</b> Loss of Required DC Power for $\geq$ 15 Minutes or Loss of Power to All UPS Busses. <i>Op. Modes: Cold Shutdown, Refueling, Defueled</i>	<b>CU3</b> All Safety Related DC Batteries Not Being Charged for $\geq$ 30 Minutes <i>Op. Modes: Cold Shutdown, Refueling, Defueled</i>
		<b>CA4</b> Inability to Maintain Plant in Cold Shutdown. <i>Op. Modes: Cold Shutdown, Refueling</i>	<b>CU4</b> UNPLANNED Loss of Decay Heat Removal Capability With Irradiated Fuel in the RPV. <i>Op. Modes: Cold Shutdown, Refueling</i>
		<b>CA7</b> Inability to Monitor and Control the Plant for $\geq$ 15 Minutes. <i>Op. Modes: Cold Shutdown, Refueling</i>	<b>CU6</b> Loss of All On-site or Off-site Communications Capabilities. <i>Op. Modes: Cold Shutdown, Refueling, Defueled</i>
			<b>CU7</b> UNPLANNED Partial Loss of Indicating, Monitoring and Control Functions for $\geq$ 15 Minutes. <i>Op. Modes: Cold Shutdown, Refueling</i>
			<b>CU8</b> Inadvertent Criticality. <i>Op. Modes: Cold Shutdown, Refueling</i>



**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

RCS Leakage.

**Operating Mode Applicability:** **Cold Shutdown**

**Example Emergency Action Level Thresholds:**

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

AP1000

1. RCS leakage results in the inability to maintain or restore Pressurizer level above the Pressurizer Low-2 Setpoint for 15 minutes or longer.

ESBWR

1. RCS leakage results in the inability to maintain or restore RPV level greater than Level 2 setpoint on B21-NBS-LI R604A-D Wide Range for 15 minutes or longer.

**Basis:**

This IC is included as a NOUE because it is considered to be a potential degradation of the level of safety of the plant. The inability to establish and maintain level is indicative of loss of RCS inventory. Prolonged loss of RCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of RCS/RPV Inventory) or CA4 (Inability to Maintain Plant in Cold Shutdown).

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

AP1000 References:

RCS-M3-001  
PXS-M3-001  
RNS-M3-001  
GW-GL-022  
TS 3.4.7  
TS 3.5

ESBWR References:

DCD Tier 2, Section 5.2.5.1.1  
TS 3.4.2



## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

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

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

[For AP1000] If RPV level continues to decrease and reaches the RCS Hot Leg level Low 4 then escalation to CA1 would be appropriate.

EAL threshold #2 addresses conditions in the refueling mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of RPV level indication is installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

### **Developer Notes:**

ESBWR Threshold 2 – Provide the site specific Drywell Equipment or Floor Drain Sumps level indication.

#### AP1000 References:

RCS-M3C-101  
WLS-M3C-101  
WLS-M3-001  
RCS-M3-001  
PXS-M3-001

#### ESBWR References:

DCD Tier 2, Section 5.2.5.1.1  
DCD Tier 2, Section 7.3.3.2  
DCD Tier 2, Table 7.3-5

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

All Safety Related DC Batteries Not Being Charged for  $\geq 30$  Minutes.

**Operating Mode Applicability: Cold Shutdown, Refueling, Defueled**

**Example Emergency Action Level Threshold:**

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

**AP1000**

1. Busses ECS-ES-1 and ECS-ES-2 de-energized for 30 minutes or longer.

**ESBWR**

1. PIP Busses A3 and B3 de-energized for 30 minutes or longer.

**Basis:**

The off-site AC power system normally supplies power for the unit in cold shutdown, refueling, and defueled conditions. Both the normal off-site and standby on-site AC power systems are non-Class 1E with no Technical Specification requirements. All safety-related functions associated with the unit in cold shutdown and refueling are provided by the safety-related on-site Class 1E DC power systems. The Passive ALWRs have standby diesel generators that are not safety-related. Storage batteries are the safety-related power source for Class 1E electric power.

In cold shutdown, the decay heat available to raise RCS temperature during a loss of RCS water inventory or loss of decay heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Technical Specification 3.9 requires the reactor to be subcritical for greater than {48 hours – AP1000} {24 hours ESBWR} prior to the movement of irradiated fuel. The heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the RPV. The heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling.

AP1000 - The loss of normal off-site AC power and standby on-site AC power systems de-energizes the RNS pumps. However, the on-site safety-related Class 1E DC power systems are rated for 24 hours and 72 hours of service based on the most-limiting post-accident electrical load requirements for powering the passive, safety-related systems, and thus remain available for a significant time following a loss of all off-site AC power and on-site AC power. Therefore, the progression of events after a loss of RNS cooling at mid-loop caused by a loss of AC power results in a heatup, an eventual boiling off of coolant, reduction of hot leg level, and actuation of passive IRWST injection. This restores RCS water inventory using only the passive cooling systems and the on-site safety-related Class 1E DC power systems.

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

Loss of DC power potentially compromises all safety related plant systems requiring electric power. When in cold shutdown, refueling, or defueled mode the event can be classified as an Unusual Event, because of

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

the significantly reduced decay heat and lower temperatures and pressures, increasing the time to restore one of the normal off-site AC power and standby on-site AC power systems. In addition, the passive design affords additional and redundant means to remove heat passively or restore power to active components. The selection of 30 minutes was chosen for allowing sufficient time for plant personnel to attempt to establish a viable diesel generator AC power supply to Busses ECS-ES-1 and 2 for AP1000 or the PIP busses for ESBWR.

### AP1000 References:

APP-ECS-E8-001  
APP-RCS-M3-001  
APP-PXS-M3-001  
APP-RNS-M3-001  
APP-ZOS-E8-001  
TS 3.9.7

### ESBWR References:

DCD Tier 2, Figure 8.1-1  
DCD Tier 2, Section 19A.3  
DCD Tier 2, Section 19A.4.3.7  
DCD Tier 2, Table 19.2-2  
TS 3.5.5

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

UNPLANNED Loss of Decay Heat Removal Capability With Irradiated Fuel in the RPV.

**Operating Mode Applicability:** **Cold Shutdown, Refueling**

**Example Emergency Action Level Thresholds: (1 or 2)**

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

AP1000

1. UNPLANNED event results in RCS temperature exceeding 200 °F on RCS-TI-135A or -135B
2. Loss of all RCS temperature and RPV level indication for 15 minutes or longer.

ESBWR

1. UNPLANNED event results in RCS temperature exceeding 200 °F on [Site specific].
2. Loss of all RCS temperature and RPV level indication for 15 minutes or longer.

**Basis:**

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

Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Technical Specification 3.9 requires the reactor to be subcritical for greater than {48 hours –AP1000} {24 hours ESBWR} prior to the movement of irradiated fuel. The heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the RPV.

Monitoring RCS temperature and RPV level will determine if escalation to the Alert level via CA4 or CA1 will occur if required.

AP1000 - Decay heat removal is normally performed by the non-safety related RNS pumps and heat exchangers. The progression of events after a loss of RNS cooling at mid-loop results in a heatup, an eventual boiling off of coolant, reduction of hot leg level, and actuation of passive IRWST injection. This restores RCS water inventory using only the safety-related passive cooling systems. However, if an unplanned event causes the both normal decay heat removal and the passive cooling systems to be lost, then RCS temperature can exceed 200°F. In addition, even though RNS may be operating and initial RCS/RPV inventory is available, a loss of all RCS temperature and level indication prevents the operator from ensuring adequate RNS decay heat removal is occurring.

ESBWR Decay heat removal is normally performed by both trains of the nonsafety-related RWCUS/SDC, which transfers sensible heat and core decay heat load produced when the reactor is being shutdown, or is in the shutdown condition, to the Reactor Component Cooling Water System. In the PRA it is assumed that

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both trains of RWCU/SDC are running, because the time periods in which only one train is running occurs when the reactor well is flooded. Failure of one of the trains is not considered an initiating event.

Any reduction of RCS inventory to the predetermined setpoint will result in an Alert based on CA1 or CA4.

### **Developer Notes:**

ESBWR Threshold #1 – fill in the site specific instrument providing RCS temperature indication.

#### AP1000 References:

APP-RCS-M3-001  
APP-PXS-M3-001  
APP-RNS-M3-001  
APP-GW-GL-022  
TS 3.4.7  
TS 3.5

#### ESBWR References:

DCD Tier 2, Table 19.2-2  
DCD Tier 2, Section 19.A.3  
DCD Tier 2, Section 19A.4.3.7  
TS Table 1.1-1

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

Loss of All On-site or Off-site Communications Capabilities.

**Operating Mode Applicability:** **Cold Shutdown, Refueling, Defueled**

**Example Emergency Action Level Thresholds: (1 or 2)**

1. Loss of all of the following on-site communications capability affecting the ability to perform routine operations:

AP1000

- EFS
- TVS

ESBWR

- Plant Page/Party Line
- PABX
- Sound Powered Phones
- Plant Radios

2. Loss of all of the following off-site communication methods affecting the ability to perform off-site notifications:

[site specific list of communications methods]

**Basis:**

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

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

EFS and TVS are comprised of the following:

- Wireless Telephone System
- Telephone-Page System
- Sound Powered System
- Security Communication System
- Closed Circuit Television System



## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

### **Developer Notes:**

Threshold #2 – Site-specific list for off-site communications loss must encompass the loss of all means of communications with off-site authorities. This should include the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems if the dedicated phone systems can make all required offsite notifications.

#### AP1000 References:

EFS-E8-001  
TVS-J7-001

#### ESBWR References:

DCD Tier 2, Section 9.5.2

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

UNPLANNED Partial Loss of Indicating, Monitoring and Control Functions for  $\geq 15$  Minutes.

**Operating Mode Applicability:** **Cold Shutdown, Refueling**

**Example Emergency Action Level Threshold:**

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

1. UNPLANNED partial Loss of [Site specific] 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 control and indication 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. A Notification of Unusual Event level is considered appropriate for this partial loss of indication and control IC due to the inherently safer condition of the core when in the cold condition. Escalation to an Alert will be via CA7 if a complete loss of control and indication occurs. Declaration of the Alert will provide the Control Room staff with additional personnel to assist in monitoring alternative indications, manipulating equipment and restoring the systems to full capability. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems due to an inadvertent loss.

AP1000 - The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shutdown the plant, and to maintain the plant in a safe shutdown condition. The Plant Control System (PLS) includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine. The Diverse Actuation System (DAS) remains available to ensure monitoring and control capability. Loss of DAS would result in escalation to CA7.

ESBWR - The Q-DCIS provides the data processing and transmission network that encompasses the four independent and separate data multiplexing divisions 1, 2, 3, and 4, corresponding to the four divisions of safety-related electrical and I&C equipment. Total loss of Q-DCIS would result in escalation to CA7

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

### **Developer Note:**

AP1000 - Fill in PLS and PMS

ESBWR - Fill in any three Q-DCIS.

#### AP1000 References:

APP-PMS-J7-001

APP-DAS-J7-001

APP-PLS-J7-001

APP-DDS-J7-001

#### ESBWR References:

DCD Tier 2, Section 7.1.3

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

Inadvertent Criticality.

**Operating Mode Applicability:** **Cold Shutdown, Refueling**

**Example Emergency Action Level Thresholds:**

AP1000

1. An UNPLANNED sustained positive startup rate on nuclear instrumentation.

ESBWR

1. An UNPLANNED SRNM sustained positive period.

**Basis:**

This IC addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel assembly loading errors (mis-located and mis-oriented). This IC indicates a potential degradation of the level of safety of the plant, warranting a NOUE classification.

Escalation would be by Emergency Director judgment.

This condition can be identified using period monitors/startup rate monitor. The term “sustained” is used in order to allow exclusion of expected short term positive periods/startup rates from planned fuel bundle or control rod movements during core alteration. 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.

AP1000 Reference:

PMS-J4-020  
PMS-J1-003

ESBWR References:

DCD Tier 2, Section 7.2.2.1.1  
TS 3.3.1.6

**Initiating Condition -- ALERT**

Loss of RCS/RPV Inventory.

**Operating Mode Applicability:**

**Cold Shutdown, Refueling**

**Example Emergency Action Level Thresholds: (1 or 2)**

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

AP1000

1. a. Pressurizer level at 12% and lowering on RCS-LT-200

**OR**

- b. RCS Hot Leg level is at Low 4 and lowering as indicated on RCS-LT-160A OR LT-160B
2. RCS level cannot be monitored for 30 minutes or longer with a loss of RCS/RPV inventory as indicated by unexplained rise in Containment sump level on WLS-LICR-034, WLS-LICR-035, OR WLS-LICR-036.

ESBWR

1. RCS inventory reduced below Level 1 setpoint on RPV Water Level B21-NBS-LI R604A-D Wide Range for 15 minutes or longer.
2. RPV level cannot be determined for 30 minutes or longer with a loss of RPV inventory as indicated by unexplained Drywell Equipment or Floor Drain Sumps level rise on [Site specific].

**Basis:**

These thresholds serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level lowering and potential core uncover. This condition will result in a minimum classification of Alert. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

AP1000 - The RCS Pressurizer level and Hot Leg level setpoints were chosen to indicate that actions must be taken to prevent reaching a level that would cause a loss of RNS cooling. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier. The pressurizer level setpoint is 12%, which is the pressurizer level low-2 setpoint. This provides CMT actuation for Core Heat Removal. The hot leg level setpoint is the hot leg level Low-4 setpoint. This activates ADS 4 and IRWST injection for Core Heat Removal.

ESBWR – The Level 1 actuation setpoint was chosen for threshold #1 to indicate that those makeup efforts are failing. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Technical Specification 3.9 requires the reactor to be subcritical for greater than {48 hours –AP1000} {24 hours ESBWR} prior to the movement of irradiated fuel. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the RPV.

If all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump or tank level changes. Sump or tank level rise must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

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

If RPV level continues to decrease then escalation to Site Area Emergency will be via CS1 (Loss of RPV Inventory Affecting Core Decay Heat Removal Capability).

### **Developer Notes:**

ESBWR Threshold #2 – fill in the site specific instrument used to determine Drywell Equipment and Floor Drain Sump levels.

#### AP1000 References:

RCS-M3 -101  
WLS-M3C-101  
WLS-M3-001  
RCS-M3-001  
PXS-M3-001

#### ESBWR References:

DCD Tier 2, Section 7.3.3.2  
DCD Tier 2, Table 7.3-5  
NEDO-33201, Section 16.3

**Initiating Condition - ALERT**

Loss of all Off-site and all On-Site AC power to Busses ECS-ES-1 and 2 (AP1000) PIP Busses (ESBWR) for 60 minutes or longer.

**Operating Mode Applicability:** Cold Shutdown, Refueling, Defueled

**Example Emergency Action Level Threshold:**

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

AP1000

1. Loss of all Off-Site and all On-Site AC Power to Busses ECS-ES-1 and 2 for the condition duration:

<b>Table: RCS Loss of Forced Cooling Thresholds</b>	
<b>RCS</b>	<b>Duration</b>
Intact but not RCS Reduced Inventory	120 minutes
Open or RCS Reduced Inventory	60 minutes

ESBWR

1. Loss of all Off-Site and all On-Site AC Power to PIP Busses A3 and B3 for the condition duration:

<b>Table: RCS Loss of Forced Cooling Thresholds</b>	
<b>RCS</b>	<b>Duration</b>
Intact	120 minutes
Open	60 minutes

**Basis:**

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

The event can be classified as an Alert when in cold shutdown, refueling, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the busses.

The 60 minute time frame when the RCS is open and 120 minute time frame when the RCS is intact were chosen based on analysis that indicates that prior to this time no challenges to passive cooling systems exist. Therefore, the times selected as the thresholds allow additional time for recovery of AC power prior to escalation to Alert.

Escalating to Site Area Emergency, if appropriate, is by Abnormal Radiation Levels / Radiological Effluent ICs.

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

### AP1000 References:

APP-GW-GL-022, AP1000 Probabilistic Risk Assessment, Chapter 54  
APP-GW-GL-026, AP1000 Implementation of the Regulatory Treatment of Nonsafety Related Systems Process.  
APP-SSAR-GSC-533, AP1000 Loss of AC Power for DCD  
APP-SSAR-GSC-536, AP1000 Safe Shutdown Temperature Evaluation  
APP-GW-GL-700, AP1000 Design Control Document, Section 16.1, Technical Specifications

### ESBWR References:

NEDO-33201, Section 16.3



**Initiating Condition -- ALERT**

Loss of Required DC Power for  $\geq 15$  Minutes or Loss of Power to All UPS Busses.

**Operating Mode Applicability:** **Cold Shutdown, Refueling, Defueled**

**Example Emergency Action Level Threshold: (1 or 2)**

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

AP1000

1. a. Loss of Required DC System Power based on less than 210VDC for ALL of the following DC switchboards
  - Division A 24-Hour Switchboard IDSA-DS-1
  - Division B 24-Hour Switchboard IDSB-DS-1
  - Division B 72-Hour Switchboard IDSB-DS-2
  - Division C 24-Hour Switchboard IDSC-DS-1
  - Division C 72-Hour Switchboard IDSC-DS-2
  - Division D 24-Hour Switchboard IDSD-DS-1

**AND**

- b. Failure to restore power to at least one required bus in less than 15 minutes from the time of loss.
2. Loss of Power to All UPS Busses.

ESBWR

1. a. Loss of All Vital DC Busses 11, 12, 21, 22, 31, 32, 41, AND 42 based on bus voltage less than 210VDC.

**AND**

- b. Failure to restore power to at least one required DC bus in less than 15 minutes from the time of loss.
2. Loss of Power to All UPS Busses.

**Basis:**

The purpose of this IC and its associated EALs is to recognize a loss of the Class 1E DC, which provides electrical power for safety related and vital control and monitoring instrumentation loads. It also provides power for safe shutdown when all the on-site and off-site AC power sources are lost and cannot be recovered for 72 hours. Loss of the vital AC instrumentation and control busses potentially compromises the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

Routinely maintenance on a division related basis is performed during shutdown periods. It is intended that the loss of the operating (operable) division is to be considered. If this loss results in the inability to maintain cold shutdown, then the Alert criteria of CA4 - Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV would also be met in addition to the CA3 Alert criteria.

Threshold 1 setpoint and Bus voltage of 210 VDC is the minimum bus voltage necessary for the operation of safety-related instrumentation and controls. This voltage value incorporates a margin significantly longer than the allowed 15 minutes of operation before the onset of inability to operate those loads.

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

### AP1000 References:

IDS-E8-001  
APP-GW-GL-022, AP1000 Probabilistic Risk Assessment, Chapter 54  
APP-GW-GL-026, AP1000 Implementation of the Regulatory Treatment of Non-safety Related Systems Process.  
APP-SSAR-GSC-533, AP1000 Loss of AC Power for DCD  
APP-SSAR-GSC-536, AP1000 Safe Shutdown Temperature Evaluation  
APP-GW-GL-700, AP1000 Design Control Document, Section 16.1, Technical Specifications

### ESBWR References:

DCD Tier 2, Figure 8.1-3  
DCD Tier 2, Section 8.3.2.1  
DCD Tier 2, Section 8.3.1.1.3  
DCD Tier 2, Figure 8.1.4  
TS Bases 3.8.2  
NEDO-33201, Section 16.3

**Initiating Condition -- ALERT**

Inability to Maintain Plant in Cold Shutdown.

**Operating Mode Applicability: Cold Shutdown, Refueling**

**Example Emergency Action Level Thresholds: (1 or 2)**

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

AP1000

1. An UNPLANNED event results in RCS Temperature greater than 200 °F as indicated on RCS-TI-135A OR -135B for greater than the specified duration on table:

<b>Table: RCS Reheat Duration Thresholds</b>		
<b>RCS</b>	<b>Containment Closure</b>	<b>Duration</b>
Intact but not RCS Reduced Inventory	N/A	60 minutes*
Open or RCS Reduced Inventory	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

2. An UNPLANNED event results in RCS Pressure increase greater than 10 psi due to a loss of RCS cooling. (This does not apply in Solid Plant conditions.)

ESBWR

1. An UNPLANNED event results in RCS temperature exceeding 200 degrees F as indicated by any of the following for greater than the specified duration on table:
  - Core Inlet Temperature on [Site specific]
  - RWCU Bottom Head Suction Temperature on [Site specific]
  - RWCU Suction Temperature on [Site specific]

<b>Table: RCS Reheat Duration Thresholds</b>		
<b>RCS</b>	<b>REACTOR BUILDING ISOLATION</b>	<b>Duration</b>
Intact	N/A	60 minutes*
Open	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

2. An UNPLANNED event results in RCS Pressure increase greater than 10 psi due to a loss of RCS cooling.

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

### **Basis:**

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

The RCS Reheat Duration Threshold table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is reduced in the AP1000. As discussed above, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed later in this basis) and is believed to be conservative given that a low pressure Containment barrier to fission product release is established.

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

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

In EAL Threshold 2, the 10 psi pressure increase covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes.

Escalation to Site Area would be via CS1 should boiling result in significant RPV level loss leading to core uncover.

AP1000 - this IC and its associated EALs are based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncover can occur. NRC analyses show that there are sequences that can cause core uncover in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

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

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

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

### **Developer Notes:**

ESBWR Threshold #1 – Fill in the site specific instruments for temperature indication.

#### AP1000 References:

RCS-M3C-101  
RCS-M3-001  
PXS-M3-001  
RNS-M3-001

#### ESBWR References:

DCD Tier 2, Section 7.7.1.2.2  
NEDO-33201, Section 16.3  
TS Table 1.1-1

**Initiating Condition -- ALERT**

Inability to Monitor and Control the Plant for  $\geq$  15 Minutes.

**Operating Mode Applicability:** Cold Shutdown, Refueling

**Example Emergency Action Level Threshold:**

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

1. UNPLANNED Loss of [Site specific] Digital Monitoring and Control Functions 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. An Alert level is considered appropriate for this IC due to the inherently safer condition of the core when in the cold condition. Declaration of the Alert will provide the Control Room staff with additional personnel to assist in monitoring alternative indications, manipulating equipment and restoring the systems to full capability. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems due to an inadvertent loss.

AP1000 - The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shutdown the plant, and to maintain the plant in a safe shutdown condition. The Plant Control System (PLS) includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine. The Diverse Actuation System (DAS) is a non-safety related system that provides a diverse backup to the protection system.

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

**Developer Note:**

AP1000 - Fill in PLS, PMS and DAS.

ESBWR - Fill in all four channels of Q-DCIS.

AP1000 References:

APP-PMS-J7-001  
APP-DAS-J7-001  
APP-PLS-J7-001  
APP-DDS-J7-001

ESBWR References:

DCD Tier 2, Section 7.1.3

**Initiating Condition -- SITE AREA EMERGENCY**

Loss of RCS/RPV Inventory Affecting Core Decay Heat Removal Capability.

**Operating Mode Applicability:** **Cold Shutdown, Refueling**

**Example Emergency Action Level Thresholds: (1 or 2 or 3)**

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

**AP1000**

1. WITH CONTAINMENT CLOSURE NOT established:

Hot Leg level less than Lo-4 on RCS LT-160A or LT-160B

2. With CONTAINMENT CLOSURE established:

RCS LT-160A or LT-160B less than 0.5%

3. RPV level cannot be determined for 60 minutes or longer with a loss of RPV inventory as indicated by:

- Containment radiation monitor reading greater than [Site specific] rad/hr on [Site specific]
- Unexplained containment sump level rise on [Site specific]
- Erratic Source Range Monitor Indication

**ESBWR**

1. WITH REACTOR BUILDING ISOLATION NOT established:

RPV level less than Level 0.5 Setpoint on B21-NBS-LI-R615A-D

2. With REACTOR BUILDING ISOLATION established:

RPV level less than Level 0 Setpoint

3. RPV level cannot be monitored for 60 minutes or longer with a loss of RPV inventory as indicated by unexplained Drywell Equipment or Floor Drain Sumps level rise on [Site specific]

**Basis:**

Under the conditions specified by this IC, continued lowering in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, pressure boundary leakage, or continued boiling in the RPV.

Escalation to a General Emergency is via CG1 or AG1.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Technical Specification 3.9 requires the reactor to be subcritical for greater than {48 hours –AP1000} {24 hours ESBWR} prior to the movement of irradiated fuel. Thus the heatup threat and therefore the threat to

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the RPV.

AP1000 – For EAL Threshold 1, the lowest observable level is used.

ESBWR – For EAL Threshold 1, the lowest level above the fuel is used.

The passive cooling systems will continue to function and provide a sufficient volume of water for cooling, therefore, 60-minute duration allows sufficient time for actions to be performed to recover needed cooling equipment and is considered to be conservative.

Declaration of a Site Area Emergency is warranted under the conditions specified by the IC due to the potential challenge to 2 fission product barriers. Escalation to a General Emergency is via CG1 (Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV) or radiological effluent IC AG1 (Off-site Dose Resulting from an Actual or IMMEDIATE Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

### **Developer Notes:**

AP1000 Threshold #3 – Fill in the calculated radiation monitor value and instrument which will have an elevated reading that corresponds to 5% fuel clad damage.

Fill in the site specific instrument used to measure containment sump level.

ESBWR Threshold #3 – Fill in the site specific instrument used to determine the Drywell Equipment and Floor Drain Sump levels.

#### AP1000 References:

APP-RCS-M3C-101  
TS 3.4.12, 3.4.13, 3.5.3,  
3.5.5 and 3.5.7

#### ESBWR References:

DCD Tier 2, Section 7.3.3.2  
DCD Tier 2, Table 7.3-5  
NEDO-33201, Section 16.3  
NEDO-33201, Table 16.3-6



**Initiating Condition -- GENERAL EMERGENCY**

Loss of RCS/RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged.

**Operating Mode Applicability:** **Cold Shutdown, Refueling**

**Example Emergency Action Level Threshold:**

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

AP1000

1. a. CONTAINMENT challenged as indicated by one or more of the following:
  - CONTAINMENT CLOSURE not established
  - Explosive mixture inside containment
  - UNPLANNED rise in containment pressure

**AND**

- b. Core uncover 30 minutes or longer as indicated by **EITHER**:
  - RCS LT-160A or LT-160B less than 0.5%.
  - CANNOT be determined with indication of core uncover by ANY of the following:
    - Containment radiation monitor reading greater than High 2 Alarm.
    - Erratic Source Range Monitor Indication
    - UNPLANNED containment sump level rise

ESBWR

1. a. CONTAINMENT challenged as indicated by one or more of the following:
  - Explosive mixture inside containment
  - UNPLANNED rise in containment pressure
  - REACTOR BUILDING ISOLATION not established
  - Reactor Building radiation monitors greater than [Site specific]

**AND**

- b. Core uncover for 30 minutes or longer as indicated by **EITHER**:
  - Less than Level 0 Setpoint.
  - RPV level cannot be monitored with core uncover indicated by EITHER of the following:
    - Drywell Radiation Monitors [Site specific] reading greater than high alarm setpoint
    - Erratic Source Range Monitor Indication
    - Unexplained Drywell Equipment or Floor Drain Sumps level rise on [Site specific]

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

### **Basis:**

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

These conditions 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.

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

If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovering time limit then escalation to GE would not occur.

Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory functions. In the early stages of a core uncovering event, it is unlikely that hydrogen buildup due to a core uncovering could result in an explosive mixture of dissolved gasses in Containment. However, Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

For ESBWR, 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.

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

As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in up-scaled radiation monitor indication and possible alarm.

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

### **Developer Notes:**

ESBWR Threshold #1.a – Fill in a calculated radiation value for Reactor Building area radiation monitors that indicates exposed fuel in the Drywell.

ESBWR Threshold #1.b – Fill in the Drywell radiation monitor instrument nomenclature.

Fill in the site specific instrument used to determine the Drywell Equipment and Floor Drain Sump levels

## COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

### AP1000 References:

APP-PXS-M3C-101  
APP-PXS-M3-001  
TS 3.4.12, 3.4.13, 3.5.3,  
3.5.5, 3.5.7 and 3.5.8

### ESBWR References:

DCD Tier 2, Section 7.3.3.2  
DCD Tier 2, Table 7.3-5  
TS 3.3.1.6  
TS 3.6.3.1  
TS Bases 3.6.1.1

ISFSI

5.7 ISFSI EALs

**Table 5-E-1: Recognition Category “E” Initiating Condition Matrix**

	<b>ALERT</b>	<b>UNUSUAL EVENT</b>
<b>Note:</b>	Security related events for ISFSIs are to be addressed under the H Series recognition category.	<b>E-HU1</b> Damage to a loaded cask CONFINEMENT BOUNDARY. <i>Op. Modes: Not Applicable</i>

**Initiating Condition – NOTIFICATION OF UNUSUAL EVENT**

Damage to a loaded cask CONFINEMENT BOUNDARY.

**Operating Mode Applicability: Not applicable**

**Example Emergency Action Level:**

1. Damage to a loaded cask CONFINEMENT BOUNDARY.

**Basis:**

A NOUE in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

The results of the ISFSI Safety Analysis Report (SAR) per NUREG 1536 or SAR referenced in the cask 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 a dropped cask, a tipped over cask, EXPLOSION, PROJECTILE damage, FIRE damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

EALs HU4 and HA4 bound security events which involve the ISFSI.



## FISSION PRODUCT BARRIERS

### 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 RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier “Loss” EALs existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier “Potential Loss” EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.
- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier.

**Table 5-F-2: ESBWR EAL Fission Product Barrier Table  
Thresholds For LOSS or POTENTIAL LOSS of Barriers\***

\*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also, multiple events could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds is IMMEDIATE. In this IMMEDIATE loss situation use judgment and classify as if the thresholds are exceeded.

<b>GENERAL EMERGENCY</b>	<b>SITE AREA EMERGENCY</b>	<b>ALERT</b>	<b>UNUSUAL EVENT</b>
Loss of ANY Two Barriers AND Loss or Potential Loss of the Third Barrier	Loss or Potential Loss of ANY Two Barriers	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	ANY Loss or ANY Potential Loss of Containment

<u>Fuel Clad Barrier Threshold Values</u>		<u>RCS Barrier Threshold Values</u>		<u>Primary Containment Barrier Threshold Values</u>	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
<b><u>1. Primary Coolant Activity Level</u></b>		<b><u>1. Primary Containment Conditions</u></b>		<b><u>1. Primary Containment Conditions</u></b>	
A. Primary coolant activity greater than 300 uCi/gm I-131	Not Applicable	A. Primary containment pressure greater than High Drywell Scram Setpoint due to RCS leakage	Not Applicable	A. Primary containment pressure rise followed by a rapid unexplained drop in primary containment pressure <b>OR</b> B. Primary containment pressure response not consistent with LOCA conditions	A. Primary containment pressure 45 psig on [Site specific]- and rising <b>OR</b> B. H <sub>2</sub> greater than 6% <b>AND</b> O <sub>2</sub> greater than 5% <b>OR</b> C. RPV pressure <b>AND</b> suppression pool temperature cannot be maintained below the HCTL
<b>OR</b>		<b>OR</b>		<b>OR</b>	
<b><u>2. Reactor Vessel Water Level</u></b>		<b><u>2. Reactor Vessel Water Level</u></b>		<b><u>2. Reactor Vessel Water Level</u></b>	
A. RPV water level cannot be restored and maintained above TAF	A. RPV water level cannot be restored and maintained above Level 0.5 Setpoint <b>OR</b> cannot be determined	A. RPV water level cannot be restored and maintained above Level 0.5 Setpoint <b>OR</b> cannot be determined	Not Applicable	Not Applicable	A. Primary Containment Flooding is required.
<b>OR</b>		<b>OR</b>		<b>OR</b>	
<b><u>3. Not Applicable</u></b>		<b><u>3. RCS Leak Rate</u></b>		<b><u>3. Primary Containment Isolation Failure or Bypass</u></b>	
Not applicable	Not applicable	A. Unisolable Main Steamline Break as indicated by: Failure of both valves in any one line to close <b>AND</b> a. Steamline High Flow <b>OR</b> b. Main Steam Line Low Pressure <b>OR</b> c. Main Steam Tunnel Ambient Temperature High	A. RCS leak greater than 100 gpm in the drywell. <b>OR</b> B. Unisolable primary system leakage outside primary containment as indicated by exceeding EITHER of the following: a. Max Normal Operating Temperature. <b>OR</b> b. Max Normal Area Radiation.	A. Failure of all valves in any one line to close <b>AND</b> direct downstream pathway to the environment exists after a primary containment isolation signal <b>OR</b> B. Intentional primary containment venting per EOPs <b>OR</b> C. Unisolable primary system leakage outside Containment as indicated by exceeding EITHER of the following: a. Max Safe Operating Temperature. <b>OR</b> b. Max Safe Area Radiation.	A. Feedline break as indicated by Main Steam tunnel Ambient Temperature High
<b>OR</b>		<b>OR</b>		<b>OR</b>	
<b><u>4. Primary Containment Radiation Monitoring</u></b>		<b><u>4. Primary Containment Radiation Monitoring</u></b>		<b><u>4. Primary Containment Radiation Monitoring</u></b>	



**Table 5-F-2: ESBWR EAL Fission Product Barrier Table  
Thresholds For LOSS or POTENTIAL LOSS of Barriers\***

\*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also, multiple events could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds is IMMEDIATE. In this IMMEDIATE loss situation use judgment and classify as if the thresholds are exceeded.

<b>GENERAL EMERGENCY</b> Loss of ANY Two Barriers AND Loss or Potential Loss of the Third Barrier	<b>SITE AREA EMERGENCY</b> Loss or Potential Loss of ANY Two Barriers	<b>ALERT</b> ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	<b>UNUSUAL EVENT</b> ANY Loss or ANY Potential Loss of Containment
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<u>Fuel Clad Barrier Threshold Values</u>		<u>RCS Barrier Threshold Values</u>		<u>Primary Containment Barrier Threshold Values</u>	
<b>LOSS</b>	<b>POTENTIAL LOSS</b>	<b>LOSS</b>	<b>POTENTIAL LOSS</b>	<b>LOSS</b>	<b>POTENTIAL LOSS</b>
A. Primary containment radiation monitor reading greater than [Site specific] R/hr on [Site specific]	Not Applicable	A. Primary containment radiation monitor reading greater than [Site specific] R/hr on [Site specific]	Not Applicable	Not applicable	A. Primary containment radiation monitor reading greater than [Site specific]R/hr on [Site specific]
<b>OR</b>		<b>OR</b>		<b>OR</b>	
<b><u>5. Other [Site specific] Indications</u></b> A. MSL Rad Monitors Drywell Fission Product Monitor – [Site specific]	[Site specific] as applicable	<b><u>5. Other [Site specific] Indications</u></b> [Site specific] as applicable	[Site specific] as applicable	<b><u>5. Other [Site specific] Indications</u></b> [Site specific] as applicable	[Site specific] as applicable
<b>OR</b>		<b>OR</b>		<b>OR</b>	
<b><u>6. Emergency Director Judgment</u></b> A. Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier		<b><u>6. Emergency Director Judgment</u></b> A. Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the RCS Barrier		<b><u>6. Emergency Director Judgment</u></b> A. Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	

## FISSION PRODUCT BARRIERS

### Basis Information For Table 5-F-2 ESBWR Emergency Action Level Fission Product Barrier Reference Table

#### **FUEL CLAD BARRIER THRESHOLDS:** (1 or 2 or 3 or 4 or 5 or 6)

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

#### **1. Primary Coolant Activity Level**

This value is 300  $\mu\text{Ci/gm}$  I-131 equivalent indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no potential loss associated with this condition.

#### **2. Reactor Vessel Water Level**

The "Loss" value is the top of active fuel which is used in EOPs to indicate challenge of core cooling. This is the minimum value to assure core cooling without further degradation of the clad.

Level 0.5 corresponds to a water level 39.4 inches above the top of the active fuel. The "Potential Loss" Threshold is the same as the RCS barrier "Loss" Threshold #2. Thus, this Threshold indicates a "Loss" of RCS barrier and a "Potential Loss" of the Fuel Clad Barrier. This Threshold appropriately escalates the emergency class to a Site Area Emergency.

#### **3. Not applicable**

#### **4. Primary Containment Radiation Monitoring**

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 greater than that specified for RCS barrier Loss Threshold #4. 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.

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

There is no potential loss associated with this condition.

#### **Developer Notes:**

The value is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300  $\mu\text{Ci/gm}$  dose equivalent I-131 or the calculated concentration equivalent to the clad damage used in Threshold #1 into the drywell atmosphere. Fill in primary containment radiation monitor number.

#### **5. Other [Site Specific] Indications**

## FISSION PRODUCT BARRIERS

This threshold addresses other [Site-specific] indications that may indicate loss or potential loss of the Fuel Clad barrier. Main Steam Line Monitors and the Drywell Fission Product Monitor at the specified values are indicative of fuel clad failure.

### **Developer Note:**

Fill in Main Steam Line Monitors and the Drywell Fission Product Monitor numbers. Fill in other instrument descriptions and instrument numbers if identified.

### **6. Emergency Director Judgment**

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

During conditions of a loss of all safety-related DC power, fission product barrier monitoring capability will be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is IMMEDIATE?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMEDIATE Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

### **RCS BARRIER THRESHOLDS: (1 or 2 or 3 or 4 or 5 or 6)**

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.

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**

Drywell pressure is based on the drywell high pressure set point which indicates a LOCA.

There is no potential loss associated with this condition.

#### **2. Reactor Vessel Water Level**

The Loss Threshold corresponds to the level which is used in EOPs to indicate challenge of core cooling. Level 0.5 corresponds to a water level above the top of the active fuel.

This Threshold appropriately escalates the emergency class to a Site Area Emergency. Thus, this Threshold indicates a loss of the RCS barrier and a Potential Loss of the Fuel Clad Barrier.

There is no potential loss associated with this condition.

#### **3. RCS Leak Rate**

## FISSION PRODUCT BARRIERS

An unisolable MSL break is a breach of the RCS barrier. Thus, this Threshold is included for consistency with the Alert emergency classification. Automatic Depressurization System automatically or manually initiated indicates a breach of the RCS.

The potential loss of RCS based on leakage is set at a level indicative of a breach of the RCS but which is well within the makeup capability of the CRD high pressure injection. Core uncover is not a significant concern for a 100 gpm leak, however, break propagation leading to significantly larger loss of inventory is possible.

Potential loss of RCS based on primary system leakage outside the drywell is determined from site-specific temperature or area radiation Max Normal setpoints in the areas of the plant which indicate a direct path from the RCS to areas outside primary containment. The indicators should be confirmed to be caused by RCS leakage. An unisolable leak which is indicated by a high alarm setpoint escalates to a Site Area Emergency when combined with Containment Barrier Threshold 3 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

#### **4. Primary Containment Radiation Monitoring**

The [Site-specific] reading is a value which indicates the release of reactor coolant to the drywell. This reading will be less than that specified for Fuel Clad Barrier Threshold #4. Thus, this threshold would be indicative of an RCS leak only. If the radiation monitor reading increased to that value specified by Fuel Clad Barrier Threshold #4, fuel damage would also be indicated.

There is no potential loss associated with this condition.

#### **Developer Notes:**

The value is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specification) into the drywell atmosphere.

Fill in primary containment radiation monitor number.

#### **5. Other [Site Specific] Indications**

This threshold addresses other [Site-specific] indications that may indicate loss or potential loss of the RCS barrier.

#### **6. Emergency Director Judgment**

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

During conditions of a loss of all safety-related DC power, fission product barrier monitoring capability will be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is IMMINENT?

## FISSION PRODUCT BARRIERS

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMEDIATE Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

### **PRIMARY CONTAINMENT BARRIER THRESHOLDS: (1 or 2 or 3 or 4 or 5 or 6)**

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

#### **1. Primary Containment Conditions**

Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of containment integrity. Drywell pressure should increase as a result of mass and energy release into containment from a LOCA. Thus, drywell pressure not increasing under these conditions indicates a loss of containment integrity. This indicator relies on the operator's recognition of an unexpected response for the condition and therefore does not have a specific value associated. The unexpected response is important because it is the indicator for a containment bypass condition.

The 45 psig for potential loss of containment is based on the primary containment design pressure. Existence of an explosive mixture means hydrogen and oxygen concentration of at least the lower deflagration limit curve exists.

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,
- 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 Note:**

Fill in primary containment pressure monitor number.

#### **2. Reactor Vessel Water Level**

There is no loss threshold associated with this condition.

The potential loss requirement for Primary Containment Flooding indicates adequate core cooling cannot be established and maintained and that core melt is possible. Entry into Primary Containment Flooding procedures is a logical escalation in response to the inability to maintain adequate core cooling.

## FISSION PRODUCT BARRIERS

Severe Accident Guidelines (SAGs) direct the operators to perform Containment Flooding when Reactor Vessel Level cannot be restored and maintained greater than a site specific value or RPV level cannot be determined with indication that core damage is occurring.

The conditions in this potential loss threshold represent potential core melt sequences which, if not corrected, could lead to vessel failure and increased potential for containment failure. In conjunction with and an escalation of the level thresholds in the Fuel and RCS barrier columns, this threshold will result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third.

### **3. Containment Isolation Failure or Bypass**

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

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in-line charcoal filter does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

Site specific EOPs may direct containment isolation valve logic(s) to be intentionally bypassed, regardless of radioactivity release rates. Under these conditions with a valid containment isolation signal, the containment should also be considered lost if containment venting is actually performed.

Intentional venting of primary containment for primary containment pressure or combustible gas control per EOPs to the secondary containment and/or the environment is considered a loss of containment. Containment venting for pressure when not in an accident situation should not be considered.

In addition, the presence of area radiation or temperature Max Safe Operating setpoints indicating unisolable primary system leakage outside the primary containment are addressed after a containment isolation. The indicators should be confirmed to be caused by RCS leakage.

There is no Potential Loss threshold associated with this item.

### **4. Primary Containment Radiation Monitoring**

There is no loss associated with this condition.

The [Site specific] reading is a value which indicates significant fuel damage well in excess of that required for loss of RCS and Fuel Clad.

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.

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

#### **Developer Note:**

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

## FISSION PRODUCT BARRIERS

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.

Fill in primary containment radiation monitor number.

### **5. Other [Site Specific] Indications**

This Threshold addresses other [Site specific] indications that may indicate loss or potential loss of the containment barrier.

#### **Developer Note:**

Fill in Main Steam Line Monitors and the Drywell Fission Product Monitor numbers. Fill in other instrument descriptions and instrument numbers if identified.

### **6. Emergency Director Judgment**

This Threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this Threshold as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification Action Statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications.

During conditions of a loss of all safety-related DC power, fission product barrier monitoring capability will be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is IMMINENT?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMINENT Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

**Table 5-F-3: AP1000 EAL Fission Product Barrier Reference Table  
Thresholds For LOSS or POTENTIAL LOSS of Barriers\***

\*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also, multiple events could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds is IMMEDIATE. In this IMMEDIATE loss situation use judgment and classify as if the thresholds are exceeded.

<b>GENERAL EMERGENCY</b> Loss of ANY Two Barriers AND Loss or Potential Loss of the Third Barrier	<b>SITE AREA EMERGENCY</b> Loss or Potential Loss of ANY Two Barriers	<b>ALERT</b> ANY Loss or ANY Potential Loss of EITHER Fuel Clad or RCS	<b>UNUSUAL EVENT</b> ANY Loss or ANY Potential Loss of Containment
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<b>Fuel Clad Barrier Threshold Values</b>		<b>RCS Barrier Threshold Values</b>		<b>Containment Barrier Threshold Values</b>	
<b>LOSS</b>	<b>POTENTIAL LOSS</b>	<b>LOSS</b>	<b>POTENTIAL LOSS</b>	<b>LOSS</b>	<b>POTENTIAL LOSS</b>
<b><u>1. Critical Safety Function Status</u></b>		<b><u>1. Critical Safety Function Status</u></b>		<b><u>1. Critical Safety Function Status</u></b>	
A. Core-Cooling Red Entry Conditions Met	A. Core Cooling-Orange Entry Conditions Met <u>OR</u> B. Heat Sink-Red Entry Conditions Met	Not Applicable	A. RCS Integrity-Red Entry Conditions Met <u>OR</u> B. Heat Sink-Red Entry Conditions Met	Not Applicable	A. Containment-Red Entry Conditions Met
<b>OR</b>		<b>OR</b>		<b>OR</b>	
<b><u>2. Primary Coolant Activity Level</u></b>		<b><u>2. RCS Leak Rate</u></b>		<b><u>2. Containment Pressure</u></b>	
A. Coolant activity greater than Dose Equivalent I-131 300 µCi/gm <u>OR</u> Xe-133 280 µCi/gm	Not Applicable	A. RCS leak rate greater than available makeup capacity as indicated by RCS subcooling less than 30 degrees on [Site specific] <u>OR</u> B. Automatic Depressurization System automatically <u>OR</u> manually initiated.	A. RCS leak rate greater than 135 gpm on [Site specific] with Letdown isolated	A. A containment pressure rise followed by a rapid unexplained drop in containment pressure. <u>OR</u> B. Containment pressure or sump level response not consistent with LOCA or MSL break conditions	A. Containment pressure 59 psig and rising on PCS-PI-012, PCS-PI-013 or PCS-PI-014 <u>OR</u> B. 4% H <sub>2</sub> on VLS-AE001, 002 or 003 <u>OR</u> C. Containment Pressure Hi/Hi Alarm on PCS-P005, PCS-006, PCS-007 or PCS-008 <u>AND</u> PCS does NOT actuate.
<b>OR</b>		<b>OR</b>		<b>OR</b>	
<b><u>3. Core Exit Thermocouple Readings</u></b>		<b><u>3. Not Applicable</u></b>		<b><u>3. Core Exit Thermocouple Reading</u></b>	
A. Core exit thermocouples greater than 1200°F	A. Core exit thermocouples greater than 700°F	Not applicable	Not applicable	Not applicable	A. Core exit thermocouples greater than 700 °F <u>AND</u> B. Restoration procedures not effective within 15 minutes
<b>OR</b>		<b>OR</b>		<b>OR</b>	



**Table 5-F-3: AP1000 EAL Fission Product Barrier Reference Table  
Thresholds For LOSS or POTENTIAL LOSS of Barriers\***

\*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also, multiple events could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds is IMMEDIATE. In this IMMEDIATE loss situation use judgment and classify as if the thresholds are exceeded.

<b>GENERAL EMERGENCY</b> Loss of ANY Two Barriers AND Loss or Potential Loss of the Third Barrier		<b>SITE AREA EMERGENCY</b> Loss or Potential Loss of ANY Two Barriers		<b>ALERT</b> ANY Loss or ANY Potential Loss of EITHER Fuel Clad or RCS		<b>UNUSUAL EVENT</b> ANY Loss or ANY Potential Loss of Containment	
<b>Fuel Clad Barrier Threshold Values</b>		<b>RCS Barrier Threshold Values</b>		<b>Containment Barrier Threshold Values</b>			
<b>LOSS</b>	<b>POTENTIAL LOSS</b>	<b>LOSS</b>	<b>POTENTIAL LOSS</b>	<b>LOSS</b>	<b>POTENTIAL LOSS</b>	<b>LOSS</b>	<b>POTENTIAL LOSS</b>
<b><u>4. Reactor Vessel Water Level</u></b>		<b><u>4. Steam Generator Tube Rupture</u></b>		<b><u>4. Steam Generator Secondary Side Release with P-to-S Leakage</u></b>			
Not Applicable	A. RCS Hot Leg Level Less than 0.5% on RCS-LT-160A or RCS-LT-160B.	A. RUPTURED Steam Generator results in a CMT/PRHR Actuation	Not Applicable	A. RUPTURED Steam Generator is also FAULTED outside of containment	Not applicable		
	<b>OR</b>		<b>OR</b>	<b><u>OR</u></b>	B. Primary-to-Secondary leakrate greater than 10 gpm AND an UNISOLABLE steam release from affected Steam Generator to the environment	<b>OR</b>	
<b><u>5. Not Applicable</u></b>		<b><u>5. Not Applicable</u></b>		<b><u>5. Containment Isolation Failure or Bypass</u></b>			
Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Failure of all valves in any one line to close <u>AND</u> direct downstream pathway to the environment exists after CMT isolation signal	Not Applicable		
	<b>OR</b>		<b>OR</b>	<b><u>OR</u></b>		<b>OR</b>	
<b><u>6. Containment Radiation Monitoring</u></b>		<b><u>6. Containment Radiation Monitoring</u></b>		<b><u>6. Containment Radiation Monitoring</u></b>			
A. Containment radiation monitor reading greater than High 1 Alarm on PXS-JE-RE-160, -161, -162, OR -163	Not Applicable	A. Containment radiation monitor reading greater than [Site specific alarm] on PXS-JE-RE-160, -161, -162, <u>OR</u> -163	Not Applicable	Not Applicable	A. Containment radiation monitor reading greater than High 2 Alarm on PXS-JE-RE-160, -161, -162, <u>OR</u> -163		
	<b>OR</b>		<b>OR</b>	<b><u>OR</u></b>		<b>OR</b>	
<b><u>7. Other [Site specific] Indications</u></b>		<b><u>7. Other [Site specific] Indications</u></b>		<b><u>7. Other [Site specific] Indications</u></b>			
[Site specific] as applicable	[Site specific] as applicable	(Site-specific) as applicable	[Site-specific] as applicable	[Site specific] as applicable	[Site specific] as applicable		
	<b>OR</b>		<b>OR</b>	<b><u>OR</u></b>		<b>OR</b>	
<b><u>8. Emergency Director Judgment</u></b>		<b><u>8. Emergency Director Judgment</u></b>		<b><u>8. Emergency Director Judgment</u></b>			
Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier		Any condition in the judgment of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier		Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier			

## FISSION PRODUCT BARRIERS

### Basis Information For Table 5-F-3 PWR Emergency Action Level Fission Product Barrier Reference Table

#### **FUEL CLAD BARRIER THRESHOLDS:** (1 or 2 or 3 or 4 or 5 or 6 or 7 or 8)

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

#### **1. Critical Safety Function Status**

These Thresholds serve as precursors to a loss of fuel clad.

Core cooling orange path indicates subcooling has been lost and that some clad damage may occur.

Core cooling red path indicated significant superheating and core uncovering and is considered to indicate a loss of the fuel clad.

Heat Sink RED when heat sink is required indicates the steam generator heat sink function is under extreme challenge and provides the potential for loss of the fuel clad.

Meeting the entry conditions satisfies these thresholds.

#### **2. Primary Coolant Activity Level**

This is a site specific value corresponding to 300  $\mu\text{Ci/gm}$  I-131 equivalent or 280  $\mu\text{Ci/gm}$  Xe-133. Assessment by the EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and the fuel barrier is considered lost.

There is no equivalent Potential Loss for this item.

#### **3. Core Exit Thermocouple Readings**

The core exit thermocouples (CETs) provide an adequate measure of core temperatures to estimate temperatures at which potential cladding damage and core over temperature may be occurring. CETs with readings greater than 700 °F indicate the onset of inadequate core cooling. Continued operation in this state can lead to a core damage sequence if Emergency Operating Procedures are not effective in restoring core cooling.

CETs with readings above 1200 °F indicate significant clad heating and the loss of the fuel clad barrier. Core exit thermocouples are included in addition to the Critical Safety Functions to include conditions when the status trees may not be in use.

#### **4. Reactor Vessel Water Level**

There is no Loss Threshold corresponding to this item.

The potential loss RCS less than 0.5% is a visible indication that corresponds to a level near the bottom of the Hot Leg.

#### **5. Not Applicable**

#### **6. Containment Radiation Monitoring**

The High 1 Alarm on PXS-JE-RE160, RE161, RE162 or RE163 is a value which indicates the release of reactor coolant, with elevated activity indicative of 5% fuel clad damage, into the containment. Use of a confirmed radiation monitoring reading can lead to an earlier Alert classification. A reactivity excursion or mechanical damage may cause fuel damage that is first detected by radiation monitors.

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.

There is no Potential Loss Threshold associated with this item.

## FISSION PRODUCT BARRIERS

### 7. **Other [Site-Specific] Indications – Not Applicable**

### 8. **Emergency Director Judgment**

The Emergency Director can declare an Alert based on the judgment that conditions exist which indicates the Loss or Potential Loss of the Fuel Cladding barrier. This can take any other factors into consideration including the inability to monitor the barrier.

During conditions of a loss of all safety-related dc power, fission product barrier monitoring capability will be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is IMMEDIATE?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMEDIATE Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

### **RCS BARRIER THRESHOLDS:** (1 or 2 or 3 or 4 or 5 or 6 or 7 or 8)

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

#### 1. **Critical Safety Function Status**

There is no Loss Threshold associated with this item.

These Thresholds serve as precursors to a loss of fuel clad.

Heat Sink RED when heat sink is required indicates the steam generator heat sink function is under extreme challenge and provides the potential for loss of the fuel clad.

An Integrity RED path indicates an extreme challenge to the safety function and a potential loss of the RCS barrier. Meeting the entry conditions satisfies these thresholds.

#### 2. **RCS Leak Rate**

The Loss Threshold addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak. Manual or automatic opening of the ADS valves initiates an RCS leak meeting the criteria.

The potential loss is based on the inability to maintain normal liquid inventory within the reactor coolant system by the Chemical and Volume Control System (CVS). Where leakage is greater than available inventory control a loss of subcooling can occur. Isolating letdown is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path such as a CVS leak exists. The intent of this condition is met if attempts to isolate Letdown are NOT successful.

#### **Developer Notes:**

Loss #2A- fill in the instrument used to determine RCS subcooling.

Potential Loss #2A – fill in the instrument used to indicate RCS leakage.

## FISSION PRODUCT BARRIERS

### **3. Not Applicable**

### **4. Steam Generator Tube Rupture (SGTR)**

A SGTR is based on the inability to maintain normal liquid inventory within the RCS by normal operation of the CVS system. The loss of the RCS barrier is based on leakage large enough to cause CMT/PRHR actuation.

By itself, this threshold will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment Barrier Loss thresholds.

There is no Potential Loss Threshold for this condition.

### **5. Not Applicable**

### **6. Containment Radiation Monitoring**

The [Site specific alarm] on PXS-JE-RE160, RE161, RE162 or RE163 is a value which indicates the release of reactor coolant to the containment with no fuel clad damage.

There is no Potential Loss Threshold associated with this item.

#### **Developer Notes:**

Fill in the alarm nomenclature for the area radiation monitors.

### **7. Other [Site-Specific] Indications – Not Applicable**

### **8. Emergency Director Judgment**

The Emergency Director can declare an Alert based on the judgment that conditions exist which indicates the Loss or Potential Loss of the RCS Barrier. This can take any other factors into consideration including the inability to monitor the barrier.

During conditions of a loss of all safety-related dc power, fission product barrier monitoring capability will be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is IMMINENT?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMINENT Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

### **CONTAINMENT BARRIER THRESHOLDS: (1 or 2 or 3 or 4 or 5 or 6 or 7 or 8)**

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. Critical Safety Function Status**

There is no Loss Threshold associated with this item.

## FISSION PRODUCT BARRIERS

A Containment RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment. Meeting the entry condition satisfies this threshold.

### **2. Containment Pressure**

Rapid unexplained loss of pressure following an initial pressure rise indicates a loss of containment integrity. Containment pressure should increase as a result of mass and energy release into the containment. In addition, containment pressure or sump level response not consistent with design basis accident conditions can also be an indicator of a Loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

Containment Pressure at 59 psig (design pressure).

The existence of an explosive mixture of hydrogen means there is potential for damage to containment.

Containment pressure at 6.2 psig (High-High setpoint) or greater indicates the pressure has reached the PCS actuation setpoint. Should the PCS system not actuate at this point, a challenge to containment that requires operator action to initiate alternate pressure suppression systems exists. This condition would represent a Potential Loss of Containment. .

### **3. Core Exit Thermocouples (CETs)**

There is no Loss threshold associated with this item.

The conditions in these thresholds represent the potential for the plant to progress to an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the Core Cooling and RCS Leakage criteria in the Fuel and RCS barrier columns, this threshold would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence.

Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

The core cooling orange path represents a potential to progress to an imminent core melt sequence, which if not corrected, could lead to vessel failure and an increased potential for containment failure. It is appropriate to allow 15 minutes for functional restoration procedures to address the core melt sequence. Whether or not the procedures will be effective should be apparent in 15 minutes. In the AP1000 plant design, there is over 546,000 gallons of water available in the IRSWT for passive injection into the RCS, which is sufficient to cover the core and flood the refueling cavity. The most severe path with respect to inventory in the AP1000 critical safety function trees is "yellow". Therefore, level considerations do not impact this EAL criterion. The establishment of this EAL threshold at greater than 700°F, therefore, also represents conservative decision-making with respect to

## FISSION PRODUCT BARRIERS

the EAL classification scheme. The increase of Core Exit Thermocouple to be at or greater than 1200°F is addressed by the EALs in the Fuel Clad column. Escalation to this EAL will result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the emergency operating procedures have been ineffective in restoring reactor vessel level above the RCS and Fuel Clad Barrier Threshold Values, there is not a "success" path and a core melt sequence is in progress.

### **4. SG Secondary Side Release With Primary To Secondary Leakage**

Steam generator tube leakage can represent the bypass of containment and the loss of the RCS barrier. This recognizes the non-isolable release path directly to the environment. The first Loss Threshold addresses the condition in which a RUPTURED steam generator is also FAULTED.

The second loss Threshold addresses SG tube leaks that exceed 10 gpm in conjunction with a UNISOLABLE release path to the environment.

### **5. Containment Isolation Failure or Bypass**

The failure of the isolation of a containment penetration allows a direct path to the environment and represents failure of the Containment barrier. The Containment barrier must be considered breached if isolation fails.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in-line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no Potential Loss threshold associated with this item.

### **6. Containment Radiation Monitoring**

There is no Loss Threshold associated with this item.

The High 2 Alarm is a value which indicates 20% fuel clad damage well in excess of the Thresholds associated with both loss of Fuel Clad and loss of RCS barriers. A major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. The High 2 Alarm on PXS-JE-RE160, RE161, RE162 or RE163 is a value which indicates the release of reactor coolant, with elevated activity indicative of 20% fuel clad damage, into the containment.

### **7. Other [Site-Specific] Indications – Not Applicable**

### **8. Emergency Director Judgment**

The Emergency Director can declare an Alert based on the judgment that conditions exist which indicates the Loss or Potential Loss of the Containment Barrier. This can take any other factors into consideration including the inability to monitor the barrier. The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification Action Statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications.

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During conditions of a loss of all safety-related dc power, fission product barrier monitoring capability will be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is IMMEDIATE?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMEDIATE Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

5.9 HAZARDS OR OTHER CONDITIONS AFFECTING PLANT SAFETY EALs

**Table 5-H-1: Recognition Category “H” Initiating Condition Matrix**

GENERAL EMERGENCY		SITE AREA EMERGENCY		ALERT		NOUE	
<b>HG1</b>	HOSTILE ACTION Resulting in Loss of Physical Control of the Facility. <i>Op. Modes: All</i>	<b>HS4</b>	HOSTILE ACTION Within the PROTECTED AREA. <i>Op. Modes: All</i>	<b>HA4</b>	HOSTILE ACTION Within the OWNER CONTROLLED AREA or Airborne Attack Threat. <i>Op. Modes: All</i>	<b>HU4</b>	Confirmed SECURITY CONDITION or Threat Which Indicates a Potential Degradation in the Level of Safety of the Plant. <i>Op. Modes: All</i>
<b>HG2</b>	Other Conditions Exist Which in the Judgment of the Emergency Director Warrant Declaration of a General Emergency. <i>Op. Modes: All</i>	<b>HS3</b>	Other Conditions Exist Which in the Judgment of the Emergency Director Warrant Declaration of a Site Area Emergency. <i>Op. Modes: All</i>	<b>HA6</b>	Other Conditions Exist Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. <i>Op. Modes: All</i>	<b>HU5</b>	Other Conditions Exist Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE. <i>Op. Modes: All</i>
		<b>HS2</b>	Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established. <i>Op. Modes: All</i>	<b>HA5</b>	Control Room Evacuation Has Been Initiated. <i>Op. Modes: All</i>		
				<b>HA1</b>	Natural or Destructive Phenomena Affecting the Plant VITAL AREAS. <i>Op. Modes: All</i>	<b>HU1</b>	Natural or Destructive Phenomena Affecting the PROTECTED AREA. <i>Op. Modes: All</i>
				<b>HA2</b>	FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown (AP1000)/Stable Shutdown (ESBWR). <i>Op. Modes: All</i>	<b>HU2</b>	FIRE Within the PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection <u>OR</u> EXPLOSION within the Protected Area Boundary. <i>Op. Modes: All</i>
				<b>HA3</b>	Access to a VITAL AREA is Prohibited Due to Release of Toxic, Corrosive, Asphyxiant or Flammable Gases Which Jeopardizes Operation of Operable Equipment Required to Maintain Safe Operations or Safely Shutdown the Reactor. <i>Op. Modes: All</i>	<b>HU3</b>	Release of Toxic, Corrosive, Asphyxiant, or Flammable Gases Deemed Detrimental to NORMAL PLANT OPERATIONS. <i>Op. Modes: All</i>





## HAZARDS OR OTHER CONDITIONS AFFECTING PLANT SAFETY

activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01g.

The AP1000 is designed for a safe shutdown earthquake (SSE) defined by a peak ground acceleration (PGA) of 0.30g. Operating Basis Earthquake (OBE) is not considered in the design basis. For the purpose of shutdown criteria the operating basis earthquake is considered to be one-third of the safe shutdown earthquake. The seismic equipment is designed to remain functional after a SSE. The time-history analyzer receives input from the triaxial acceleration sensors. It provides for initiation of audible and visual alarms in the main Control Room. Alarms are initiated when a seismic event exceeds a predetermined value or the calculated cumulative absolute velocity (CAV).

The ESBWR peak ground acceleration (PGA) of the SSE at the foundation level is 0.3g in the horizontal direction. The PGA in the vertical direction is equal to the horizontal PGA. The Operating Basis Earthquake (OBE) is not an ESBWR design requirement. Consistent with Appendix S to 10 CFR 50, the design requirements associated with the OBE, when the level of OBE ground motion is chosen to be one-third of the SSE ground motion, are satisfied without performing explicit response or design analyses. The ESBWR OBE ground motion is one-third of the SSE ground motion.

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

Threshold #2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind site specific value is based on FSAR design basis and is within the range of the instrumentation available for wind speed. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert.

Threshold #3 addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via HU2 and HU3. Generator seal damage observed after generator purge does not meet the intent of this Threshold because it did not impact normal operation of the plant. This Threshold is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by PROJECTILES generated by the failure. These events would be classified by the radiological ICs or Fission Product Barrier ICs.

Threshold #4 is other site-specific phenomena that can also be precursors of more serious events.

### **Developer Notes:**

AP1000 Threshold #2 – Fill in the instrument tag number for the wind speed indicator.

Threshold #4 - Sites subject to severe weather (i.e., hurricane, flood, or seiche) should include a Threshold based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.).

HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

AP1000 References:

APP-SJS-J7-001  
APP-RCS-M3-001  
APP-CNS-M3-001

ESBWR References:

DCD Tier 2, Section 3.7.1.1  
DCD Tier 2 Table 3G.1-2



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

The list is limited and applies to buildings and areas in actual contact with or immediately adjacent to VITAL AREAS or other significant buildings or areas. The intent of this IC is not to include buildings (i.e., warehouses) or areas that are not in actual contact with or immediately adjacent to VITAL AREAS. This excludes FIRES within administration buildings, waste-basket FIRES, and other small FIRES of no safety consequence. Immediately adjacent implies that the area immediately adjacent contains or may contain equipment or cabling that could impact equipment located in VITAL AREAS or the fire could damage equipment inside VITAL AREAS or that precludes access to VITAL AREAS.

For Threshold #2 only those EXPLOSIONS of sufficient force to damage permanent structures or equipment within the PROTECTED AREA should be considered. No attempt is made in this Threshold to assess the actual magnitude of the damage. The occurrence of the EXPLOSION is sufficient for declaration. The Emergency Director also needs to consider any security aspects of the EXPLOSION, if applicable.

Escalation to a higher emergency class is by IC HA2, "FIRE Affecting the Operability of Plant Safety Systems Required for the Current Operating Mode."

AP1000 References:

FPS-M3-001  
CNS-M3-001  
TS 5.4

ESBWR References:

DCD Tier 2, Section 1.1.2.1  
DCD Tier 2, Section 9.5.1  
TS 5.4.1



HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

**HU4**

**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 Level Thresholds: (1 or 2 or 3)**

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the [Site specific security shift supervision].
2. A credible [Site specific] security threat notification.
3. A validated notification from NRC providing information of an aircraft threat.

**Basis:**

*Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.*

Security events which do not represent a potential degradation in the level of safety of the plant, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under HA4, HS4 and HG1.

A higher initial classification could be made based upon the nature and timing of the threat and potential consequences. Consider upgrading the emergency response status and emergency classification in accordance with the Safeguards Contingency Plan and Emergency Plan.

Threshold #1 is based on [Site specific] Safeguards Contingency Plan.

Reference is made to security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.

This threshold is based on [Site specific] security plans. [Site specific] Safeguards Contingency Plans are based on guidance provided by NEI 03-12.

Threshold #2 is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of Unusual Event.

The determination of “credible” is made through use of information found in the site specific Safeguards Contingency Plan

Threshold #3 is to ensure that notifications for the security threat are made in a timely manner and that Off-site Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Only the plant to which the specific threat is made need declare the NOUE. This EAL is based on NRC Bulletin 2005-02, Emergency Preparedness And Response Actions For Security-Based Events.

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The NRC Headquarters Operations Officer will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level would be via HA4 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

**Developer Notes:**

Threshold #1 – Fill in the title of the security position providing information to the Control Room.

Threshold #2 – Fill in the name of the site.

AP1000 References:

ESBWR References:

NRC Bulletin 2005-02

NRC Bulletin 2005-02





HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

**HA1**

**Initiating Condition -- ALERT**

Natural or Destructive Phenomena Affecting the Plant VITAL AREAS.

**Operating Mode Applicability:** All

**Example Emergency Action Level Thresholds:** (1 or 2 or 3 or 4 or 5 or 6)

AP1000

1. Seismic event greater than 0.10g as indicated by the time history analyzer initiation of the Control Room alarm.

**AND**

Confirmed by **ANY** of the following:

- Earthquake felt in plant
  - National Earthquake Information Center
  - Control Room indication of degraded performance of systems required for the safe shutdown of the plant.
2. Tornado striking or high winds greater than 145 mph resulting in **VISIBLE DAMAGE** to any of the following structures containing safety systems or components **OR** Control Room indication of degraded performance of those systems.
    - Containment Building
    - Shield Building
    - Aux Building
  3. Internal flooding in any areas of the plant that creates an industrial safety hazard (e.g., electric shock) that precludes access necessary to operate or monitor equipment.
  4. Vehicle crash within **PROTECTED AREA** boundary and resulting in **VISIBLE DAMAGE** to any of the following structures containing safety systems or components **OR** Control Room indication of degraded performance of those safety systems:
    - Containment
    - Shield Building
    - Aux Building
  5. [Site-Specific] occurrences within **PROTECTED AREA** boundary and resulting in **VISIBLE DAMAGE** to plant structures containing equipment necessary for safe shutdown, or has caused damage as evidenced by Control Room indication of degraded performance of those systems.

ESBWR

1. Seismic event greater than 0.10g as indicated by seismic instrumentation.

**AND**

Confirmed by **ANY** of the following:

- Earthquake felt in plant
- National Earthquake Information Center

HAZARDS OR OTHER CONDITIONS  
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- Control Room indication of degraded performance of systems required for the safe shutdown of the plant.
2. Tornado striking or high winds greater than 150 mph resulting in VISIBLE DAMAGE to any of the following structures containing safety systems or components OR Control Room indication of degraded performance of those systems.
    - Containment
    - Reactor Building
    - Control Building
    - Fuel Building
    - Electrical Building
    - Ancillary Diesel Building
  3. Internal flooding in any areas of the plant that creates an industrial safety hazard (e.g., electric shock) that precludes access necessary to operate or monitor equipment.
  4. Turbine failure-generated PROJECTILES result in any VISIBLE DAMAGE to or penetration of the Electrical Building.
  5. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following structures containing safety systems or components OR Control Room indication of degraded performance of those safety systems:
    - Containment
    - Reactor building
    - Control Building
    - Fuel Building
    - Electrical Building
    - Ancillary Diesel Building
  6. [Site-Specific] occurrences within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to plant structures containing equipment necessary for safe shutdown, or has caused damage as evidenced by Control Room indication of degraded performance of those systems.

**Basis:**

These Thresholds escalate from HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance, represents a challenge to the FSAR design basis, or prevents operators from taking required actions to place the reactor in a safe condition. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this Threshold to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation to higher classifications occur on the basis of System Malfunctions.

HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

Threshold #1 is based on the FSAR design basis. The 0.10g value is conservatively chosen as one third of the design Safe Shutdown Earthquake setpoint of 0.30g. Seismic events of this magnitude can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. See EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, for information on seismic event categories.

Threshold #2 is based on design wind loading basis and is within the range of the instrumentation available for wind speed. Wind loads of this magnitude can cause damage to safety functions.

Threshold #s 2, 3, and 4 specifies site-specific safety structure, system, or component and functions required for safe shutdown of the plant.

Threshold #3 addresses the effect of internal flooding that has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment.

Threshold #4 addresses the threat to safety related equipment imposed by PROJECTILES generated by main turbine rotating component failures. Therefore, this EAL threshold is consistent with the definition of an ALERT in that if PROJECTILES have damaged or penetrated areas containing safety structure, system, or component the potential exists for substantial degradation of the level of safety of the plant.

For the AP1000, the turbine generator is located north of the nuclear island with its shaft oriented north-south. In this orientation the potential for damage from turbine missiles is negligible. Safety-related structures, systems, and components are located outside the high-velocity, low-trajectory missile strike zone. Thus, postulated low-trajectory missiles cannot directly strike safety-related areas. The potential for a high-trajectory missile to impact safety-related areas of the AP1000 is less than 10<sup>-7</sup>. Based on this very low probability, the potential damage from a high-trajectory missile is not evaluated. Since no credible missile damage from the turbine generator exists, it is not included as an EAL.

Threshold #5 (AP1000 #4) addresses crashes of vehicle types large enough to cause significant damage to safety structure, system, or component containing functions and systems required for safe shutdown of the plant.

Threshold #6 (AP1000 #5) is other site-specific phenomena that can also be precursors of more serious events.

**Developer Notes:**

AP1000 Threshold #5 - Sites subject to severe weather (i.e., hurricane, flood, or seiche) should include a Threshold based on damage associated with the severe weather condition.

ESBWR Threshold #6- Sites subject to severe weather (i.e., hurricane, flood, or seiche) should include a Threshold based on damage associated with the severe weather condition.

AP1000 References:

APP-SJS-J7-001  
APP-RCS-M3-001  
APP-CNS-M3-001

ESBWR References:

DCD Tier 2, Section 1.1.2.1  
DCD Tier 2, Section 3.7.1.1  
DCD Tier 2, Table 3G.1-2



HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

This EAL specifies structures or areas that contain safety system, or component and functions required for safe shutdown of the plant. Site specific Safe Shutdown Analysis was consulted for equipment and plant areas required to establish or maintain safe shutdown.

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

AP1000 References:

APP-RCS-M3-001  
APP-CNS-M3-001  
APP-FPS-M3-001  
APP-GW-GJP-305

ESBWR References:

DCD Tier 2, Chapter 1, Section 1.1.2.1  
DCD Tier 2, Chapter 9, Section 9.5.1

HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

**HA3**

**Initiating Condition -- ALERT**

Access to a VITAL AREA is Prohibited Due to Release of Toxic, Corrosive, Asphyxiant or Flammable Gases Which Jeopardizes Operation of Operable Equipment Required to Maintain Safe Operations or Safely Shutdown the Reactor.

**Operating Mode Applicability:** **All**

**Example Emergency Action Level Thresholds**

*Note: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.*

1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant, or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.

**Basis:**

Gases in a VITAL AREA can affect the ability to safely operate or safely shutdown the reactor.

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

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

An uncontrolled release of flammable gases within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gases, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gases which can ignite/support combustion.

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

AP1000 References:

None

ESBWR References:

DCD Tier 2, Chapter 3, Section 3.1.2.10

HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

**HA4**

**Initiating Condition - ALERT**

HOSTILE ACTION Within the OWNER CONTROLLED AREA or Airborne Attack Threat.

**Operating Mode Applicability: All**

**Example Emergency Action Level Threshold: (1 or 2)**

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the [Site specific security shift supervision].
2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

**Basis:**

*Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.*

These EALs address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

Threshold #1 addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OCA. Those events are adequately addressed by other EALs.

Although nuclear plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for OROs to be notified and encouraged to begin activation (if they do not normally) to be better prepared should it be necessary to consider further actions.

If not previously notified by the NRC that the airborne HOSTILE ACTION was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

Threshold #2 addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This EAL is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.



HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

**Developer Notes:**

Threshold #1 – Fill in the site specific security shift supervision title.

AP1000 References:

ESBWR References:

NRC Bulletin 2005-02

NRC Bulletin 2005-02







HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

**HS3**

**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 Level Threshold:**

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 class description for Site Area Emergency.

AP1000 References:

None

ESBWR References:

None

HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

**HS4**

**Initiating Condition – - SITE AREA EMERGENCY**

HOSTILE ACTION Within the PROTECTED AREA.

**Operating Mode Applicability: All**

**Example Emergency Action Level Threshold:**

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the [Site specific security shift supervision].

**Basis:**

This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.

Although nuclear plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for OROs to be notified and encouraged to begin preparations for public protective actions (if they do not normally) to be better prepared should it be necessary to consider further actions.

If not previously notified by NRC that the airborne HOSTILE ACTION was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

**Developer Notes:**

Threshold #1 – Fill in the site specific security shift supervision title.

HAZARDS OR OTHER CONDITIONS  
AFFECTING PLANT SAFETY

AP1000 References:

NRC Bulletin 2005-02

ESBWR References:

NRC Bulletin 2005-02







## SYSTEM MALFUNCTIONS

### 5.10 SYSTEM MALFUNCTION EALs

**Table 5-S-1: Recognition Category “S” Initiating Condition Matrix**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
	<p><b>SS1</b> Loss of Required DC Power for <math>\geq 15</math> Minutes or Loss of Power to All UPS Busses. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p>	<p><b>SA1</b> All Safety Related DC Batteries Not Being Charged for <math>\geq 60</math> Minutes. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p>	<p><b>SU1</b> All Safety Related DC Batteries Not Being Charged for <math>\geq 30</math> Minutes. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p>
<p><b>SG2</b> Automatic Scram (Trip) and All Manual Actions Fail to Shutdown the Reactor and Indication of an Extreme Challenge to the Ability to Cool the Core Exists. <i>Op. Modes: Power Operation, Startup</i></p>	<p><b>SS2</b> Automatic Scram (Trip) Fails to Shutdown the Reactor AND Manual Actions Taken from the Reactor Control Console are NOT Successful in Shutting Down the Reactor. <i>Op. Modes: Power Operation, Startup</i></p>	<p><b>SA2</b> Automatic Scram (Trip) Fails to Shutdown the Reactor AND the Manual Actions Taken from the Reactor Control Console are Successful in Shutting Down the Reactor. <i>Op. Modes: Power Operation, Startup</i></p>	<p><b>SU2</b> Inability to Reach Required Shutdown Mode Within Technical Specification Limits. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p> <p><b>SU4</b> Fuel Clad Degradation. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p> <p><b>SU5</b> RCS Leakage. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p> <p><b>SU6</b> Loss of All On-site or Off-site Communications Capabilities. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p>
	<p><b>SS7</b> Inability to Monitor and Control the Plant for <math>\geq 15</math> Minutes. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p>	<p><b>SA7</b> UNPLANNED Partial Loss of Indicating, Monitoring and Control Functions for <math>\geq 15</math> Minutes. <i>Op. Modes: Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown</i></p>	<p><b>SU8</b> Inadvertent Criticality. <i>Op. Modes: Hot Standby/Shutdown, Safe/Stable Shutdown</i></p>

## SYSTEM MALFUNCTION

SU1

### **Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

All Safety Related DC Batteries Not Being Charged for  $\geq 30$  Minutes.

#### **Operating Mode Applicability:**

**Power Operation, Startup, Hot  
Standby/Shutdown, Safe/Stable Shutdown**

Example Emergency Action Level Threshold:

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

#### **AP1000**

1. Busses ECS-ES-1 and ECS-ES-2 de-energized for 30 minutes or longer.

#### **ESBWR**

1. PIP Busses A3 and B3 de-energized for 30 minutes or longer.

#### **Basis:**

Prolonged de-energization of the busses reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of DC Power. 30 minutes was selected as a threshold to exclude transient or momentary losses of AC power.

The Passive ALWRs have standby diesel generators that are not safety-related. Storage batteries are the safety-related power source for Class 1E electric power.

#### **AP1000 References:**

APP-ECS-E8-001  
APP-ZOS-E8-001  
TS 3.8

#### **ESBWR References:**

DCD Tier 2, Section 1.2.2.13.1  
DCD Tier 2, Section 8.1.5.1  
DCD Tier 2, Section 8.1.5.2  
DCD Tier 2, Figure 8.1-1  
DCD Tier 2, Section 15.5.5  
TS 3.8

SYSTEM MALFUNCTION

**SU2**

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

Inability to Reach Required Shutdown Mode Within Technical Specification Limits.

**Operating Mode Applicability:**

**Power Operation, Startup, Hot  
Standby/Shutdown, Safe/Stable Shutdown**

**Example Emergency Action Level Threshold:**

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

**Basis:**

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. The initiation of plant shutdown required by the site Technical Specifications requires a four hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable Action Statement time in the Technical Specifications. An immediate NOUE is required when the plant is not brought to the required operating mode within the allowable Action Statement time in the Technical Specifications. Declaration of a NOUE is based on the time at which the LCO-specified Action Statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

AP1000 Reference:

ESBWR References:

TS 3.0.3

TS 3.0.3

## SYSTEM MALFUNCTION

SU4

### **Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

Fuel Clad Degradation.

**Operating Mode Applicability:**

**Power Operation, Startup, Hot  
Standby/Shutdown, Safe/Stable Shutdown**

### **Example Emergency Action Level Thresholds: (1 or 2)**

AP1000

1. Liquid Sample Radiation Monitor PSS-RICA-050 High Alarm indicating fuel clad degradation greater than Technical Specification 3.4.10 allowable limits.
2. Dose equivalent I-131 greater than 60  $\mu\text{Ci/gm}$  OR dose equivalent Xe-133 greater than 280  $\mu\text{Ci/gm}$  for more than 6 hours from sampling and analysis.

ESBWR

1. [Site specific Radiation Monitor] readings indicating fuel clad degradation greater than Technical Specification 3.4.3 allowable limits.
2. Dose equivalent I-131 greater than 0.2 $\mu\text{Ci/gm}$  for more than 48 hours or greater than 4.0 $\mu\text{Ci/gm}$  indicating fuel clad degradation greater than Technical Specification 3.4.3 allowable limits.

### **Basis:**

This IC is included as a NOUE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. EAL #1 addresses site-specific radiation monitor readings such as BWR air ejector monitors, PWR failed fuel monitors, etc., that provide indication of fuel clad integrity. EAL #2 addresses coolant samples exceeding coolant Technical Specifications for iodine spike. Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs.

AP1000 - PSS-RICA-050, provides early indication of significant increase in radioactivity of the reactor coolant, indicating a possible fuel cladding breach. On high alarm, the primary sampling system liquid sample radiation monitor isolates the sample flow by closing the outside containment isolation valve (PSS-PL-V011) and initiates an alarm in the main Control Room and locally to alert the operator. At that time, sampling and analysis would be performed to verify compliance with the Technical Specification 3.4.10 RCS Specific Activity limits.

Technical Specification 3.4.10 limits the allowable concentration of iodines and noble gases, such as xenon, in the reactor coolant. Limiting Condition for Operation (LCO) limits are established to be consistent with fuel defect level of 0.25 percent and to ensure that plant operation remains within conditions assumed for shielding and DBA release analyses.

Technical Specification Surveillance Requirement (SR) 3.4.10.1 requires performing a measure of the noble gas specific activity of the reactor coolant once every 7 days, which provides an indication of any increase in the release of noble gas activity from fuel rods containing cladding defects. SR 3.4.10.2 requires performing a measure of the iodine specific activity of the reactor coolant once every 14 days, and between 2 to 6 hours after a reactor power increase of greater than or equal to 15% of Rated Thermal Power within a 1 hour

## SYSTEM MALFUNCTION

period. Trending the results of these surveillances allows proper remedial action to be taken PRIOR to reaching the LCO upper limits under normal operating conditions.

ESBWR – The RWCU/SDC System line break outside containment release is the bounding accident with respect to offsite doses. The limits on the specific activity of the primary coolant ensure that the 2 hour Total Effective Dose Equivalent (TEDE) doses at the site boundary, resulting from an RWCU/SDC System line break outside containment during steady state operations, will not exceed the dose guidelines of Regulatory Guide 1.183.

The specific iodine activity is limited to  $\leq 0.2 \mu\text{Ci/gm}$  Dose Equivalent. This limit ensures the source term assumed in the safety analysis is not exceeded, so any release of radioactivity to the environment during an RWCU/SDC System line break outside containment is less than the Regulatory Guide 1.183 limits.

TS 3.4.3 allows the specific iodine activity to be greater than  $0.2 \mu\text{Ci/gm}$  but less than  $4.0 \mu\text{Ci/gm}$  for 48 hours to accommodate crud burst clean-up.

### **Developer Note:**

ESBWR Threshold #1 – Fill in the site specific radiation monitor number.

AP1000 References:

APP-PSS-M3C-101  
TS 3.4.10

ESBWR References:

TS 3.4.3

SYSTEM MALFUNCTION

**SU5**

**Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

RCS Leakage.

**Operating Mode Applicability:**

**Power Operation, Startup, Hot  
Standby/Shutdown, Safe/Stable Shutdown**

**Example Emergency Action Level Thresholds: (1 or 2)**

AP1000

1. Unidentified leakage greater than 5 gpm.
2. Identified leakage greater than 25 gpm.

ESBWR

1. Unidentified or pressure boundary leakage greater than 50 gpm.
2. Total leakage greater than 75 gpm.

**Basis:**

This IC is included as a NOUE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. The value for the unidentified leakage (including the pressure boundary) was selected as it is observable with normal Control Room indications and is 10 times the Technical Specification limit. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation affecting primary inventory should be excluded from this IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC if the relief valve cannot be isolated. ADS valves are not considered to be relief valves when functioning as designed. Leakage through the ADS valves should be compared against the threshold for RCS leakage applicability.

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 and is 2.5 times the Technical Specification limit. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs.

AP1000 References:

ESBWR References:

TS 3.4.7

TS 3.4.2

## SYSTEM MALFUNCTION

**SU6**

### **Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

Loss of All On-site or Off-site Communications Capabilities.

#### **Operating Mode Applicability:**

**Power Operation, Startup, Hot  
Standby/Shutdown, Safe/Stable Shutdown**

#### **Example Emergency Action Level Thresholds: (1 or 2)**

1. Loss of all of the following on-site communication methods affecting the ability to perform routine operations:

AP1000

- EFS
- TVS

ESBWR

- Plant Page/Party Line
- PABX
- Sound Powered Phones
- Plant Radios

2. Loss of all of the following off-site communication methods affecting the ability to perform offsite notifications:

[Site specific list of communications methods]

#### **Basis:**

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

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

- Wireless Telephone System
- Telephone-Page System
- Sound Powered System
- Security Communication System
- Closed Circuit Television System



## SYSTEM MALFUNCTION

### **Developer Notes:**

Threshold #2 – Site-specific list for off-site communications loss must encompass the loss of all means of communications with off-site authorities. This should include the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems, if the dedicated phone systems can make all required offsite notifications.

#### AP1000 References:

APP-EFS-J7-001  
APP-TVS-J7-001

#### ESBWR References:

DCD Tier 2, Section 9.5.2

## SYSTEM MALFUNCTION

**SU8**

### **Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT**

Inadvertent Criticality.

**OPERATING MODE APPLICABILITY**      **Hot Standby/Shutdown, Safe/Stable Shutdown**

#### **Example Emergency Action Level Threshold:**

AP1000

1. UNPLANNED sustained positive startup rate on nuclear instrumentation.

ESBWR

1. UNPLANNED SRNM sustained positive period.

#### **Basis:**

This IC addresses inadvertent criticality events. This IC indicates a potential degradation of the level of safety of the plant, warranting a NOUE classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling IC is CU8.

This condition can be identified using period monitors/startup rate monitor. The term “sustained” is used in order to allow exclusion of expected short term positive periods/startup rates from planned fuel bundle or control rod movements during core alteration. 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 Matrix, as appropriate to the operating mode at the time of the event, or by Emergency Director judgment.

AP1000 References:

APP-PMS-J1-003

ESBWR References:

TS 3.3.1.6

DCD Tier 2 Section 7.2.2.1.1

## SYSTEM MALFUNCTION

**SA1**

### **Initiating Condition -- ALERT**

All Safety Related DC Batteries Not Being Charged for  $\geq 60$  Minutes.

### **Operating Mode Applicability:**

**Power Operation, Startup, Hot  
Standby/Shutdown Safe/Stable Shutdown**

### **Example Emergency Action Level Threshold:**

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

#### **AP1000**

1. Busses ECS-ES-1 and ECS-ES-2 de-energized for 60 minutes or longer.

#### **ESBWR**

1. PIP Busses A3 and B3 de-energized for 60 minutes or longer.

### **Basis:**

This IC and the associated Threshold is intended to provide an escalation from IC SU1. Prolonged de-energization of the busses reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of DC Power. No operator action is required for 72 hours. 60 minutes was selected as an escalation to allow for ERO augmentation support to the operating crew to be initiated.

The condition indicated by this IC is the degradation of the AC power systems.

There are no safety-related functions with respect to off-site or on-site AC power (excluding the 120V Vital AC power system supplied from the batteries powering the Safety Related DC busses through inverters) in the advanced passive plant design that are required for the protection of any of the fission product barriers. All electrical power requirements that are necessary to protect the health and safety of the public and the fission product barriers are part of the DC power system design which is completely independent of the off-site or on-site AC power systems.

#### **AP1000 References:**

APP-ECS-E8-001  
APP-EDS-E8-001  
APP-IDS-E8-001  
TS 3.8

#### **ESBWR References:**

DCD Tier 2, Section 1.2.2.13.1  
DCD Tier 2, Section 8.1.5.1  
DCD Tier 2, Section 8.1.5.2  
DCD Tier 2, Figure 8.1-1  
DCD Tier 2, Section 15.5.5  
TS 3.8



## SYSTEM MALFUNCTION

### AP1000 References:

APP-PMS-J7-001  
APP-DAS-J7-001  
APP-PLS-J7-001  
APP-RCS-M3-001  
TS 3.3.1

### ESBWR References:

DCD Tier 2, Table 7.2-4

## SYSTEM MALFUNCTION

**SA7**

### **Initiating Condition -- ALERT**

UNPLANNED Partial Loss of Indicating, Monitoring and Control Functions for  $\geq 15$  Minutes.

### **Operating Mode Applicability:**

**Power Operation, Startup, Hot Standby/Shutdown, Safe/Stable Shutdown**

### **Example Emergency Action Level Threshold:**

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

1. UNPLANNED partial Loss of [Site specific] 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 control and indication 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. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems due to an inadvertent loss.

AP1000 - The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shutdown the plant, and to maintain the plant in a safe shutdown condition. The Plant Control System (PLS) includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine. The Diverse Actuation System (DAS) remains available to ensure monitoring and control capability. Loss of DAS would result in escalation to SS7 due to the operating crew being unable monitor and control the plant.

ESBWR - The Q-DCIS provides the data processing and transmission network that encompasses the four independent and separate data multiplexing divisions 1, 2, 3, and 4, corresponding to the four divisions of safety-related electrical and I&C equipment. Total loss of Q-DCIS would result in escalation to SS7 due to the operating crew being unable monitor and control the plant.

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### **Developer Note:**

AP1000 - Fill in PLS and PMS

ESBWR - Fill in any three Q-DCIS.

AP1000 References:

APP-PMS-J7-001

APP-DAS-J7-001

APP-PLS-J7-001

APP-DDS-J7-001

ESBWR References:

DCD Tier 2, Section 7.1.3

## SYSTEM MALFUNCTION

**SS1**

### **Initiating Condition -- SITE AREA EMERGENCY**

Loss of Required DC Power for  $\geq$  15 Minutes or Loss of Power to All UPS Busses.

#### **Operating Mode Applicability:**

**Power Operation, Startup, Hot  
Standby/Shutdown, Safe/Stable Shutdown**

#### **Example Emergency Action Level Threshold: (1 or 2)**

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

#### **AP1000**

1. a. Loss of Required DC System Power based on less than 210VDC for ALL of the following DC switchboards
  - Division A 24-Hour Switchboard IDSA-DS-1
  - Division B 24-Hour Switchboard IDSB-DS-1
  - Division B 72-Hour Switchboard IDSB-DS-2
  - Division C 24-Hour Switchboard IDSC-DS-1
  - Division C 72-Hour Switchboard IDSC-DS-2
  - Division D 24-Hour Switchboard IDSD-DS-1

#### **AND**

- b. Failure to restore power to at least one required bus in less than 15 minutes from the time of loss.
2. Loss of Power to All UPS busses.

#### **ESBWR**

1. a. Loss of All Vital DC Busses 11, 12, 21, 22, 31, 32, 41, AND 42 based on bus voltage less than 210VDC.

#### **AND**

- b. Failure to restore power to at least one required DC bus in less than 15 minutes from the time of loss.
2. Loss of Power to All UPS busses.

#### **Basis:**

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

The purpose of this IC and its associated EALs is to recognize a loss of the Class 1E DC, which provides electrical power for safety related and vital control and monitoring instrumentation loads. It also provides power for safe shutdown when all the on-site and off-site AC power sources are lost and cannot be recovered



## SYSTEM MALFUNCTION

for 72 hours. Loss of the vital AC instrumentation and control busses potentially compromises the ability to monitor and control the plant. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

Threshold 1 setpoint and Bus voltage of 210 VDC is the minimum bus voltage necessary for the operation of safety-related instrumentation and controls. Fifteen minutes for the initiating condition was selected as a threshold to exclude transient or momentary power losses. This voltage value incorporates a margin significantly longer than the allowed 15 minutes of operation before the onset of inability to operate those loads.

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

Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation, or Emergency Director judgment ICs.

### AP1000 References:

APP-ECS-E8-001  
APP-EDS-E8-001  
APP-IDS-E8-001  
TS 3.8

### ESBWR References:

DCD Tier 2, Section 8.3.2.1  
DCD Tier 2, Figure 8.1-3  
DCD Tier 2, Section 8.3.1.1.3  
DCD Tier 2, Figure 8.1.4  
TS Bases 3.8.1



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AP1000 References:

APP-PMS-J7-001  
APP-DAS-J7-001

ESBWR References:

DCD Tier 2, Table 7.2.4

## SYSTEM MALFUNCTION

SS7

### **Initiating Condition -- SITE AREA EMERGENCY**

Inability to Monitor and Control the Plant for  $\geq$  15 Minutes.

#### **Operating Mode Applicability:**

**Power Operation, Startup, Hot  
Standby/Shutdown, Safe/Stable Shutdown**

#### **Example Emergency Action Level Threshold:**

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

1. UNPLANNED Loss of [Site specific] Digital Monitoring and Control Functions 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. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems due to an inadvertent loss.

AP1000 - The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shutdown the plant, and to maintain the plant in a safe shutdown condition. The Plant Control System (PLS) includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine. The Diverse Actuation System (DAS) is a non-safety related system that provides a diverse backup to the protection system.

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

#### **Developer Note:**

AP1000 - Fill in PLS, PMS and DAS.

ESBWR - Fill in all four channels of Q-DCIS.

#### AP1000 References:

APP-PMS-J7-001  
APP-DAS-J7-001  
APP-PLS-J7-001  
APP-DDS-J7-001

#### ESBWR References:

DCD Tier 2, Section 7.1.3



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For ESBWR, the extreme challenge to the ability to cool the core is intended to mean that the reactor vessel water level cannot be restored and maintained above the Top of Active Fuel (TAF) as described in the EOP bases.

Another consideration is the inability to initially remove heat during the early stages of this sequence. For AP1000, if emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist. This EAL equates to a Heat Sink RED condition combined with a Subcriticality RED condition.

For ESBWR, considerations include inability to remove heat via the main condenser, or via the suppression pool.

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.
- 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. Therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the Safety System Design, a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum off-site intervention time.

### AP1000 References:

APP-PMS-J4-020  
APP-PMS-J7-001  
APP-DAS-J7-001  
APP-PLS-J7-001

### ESBWR References:

DCD Tier 2, Section 15.5.4

## **Appendix A**

### Basis for Radiological Effluent Initiating Conditions

#### **INTRODUCTION – APPENDIX A – BASIS FOR RADIOLOGICAL EFFLUENT ICs**

This appendix supplements the basis information provided in Section 5 for initiating conditions AU1, AA1, AS1, and AG1.

This appendix will be structured into seven major sections. They are:

1. Purpose of the effluent ICs/EALs and their relationship to other ICs/EALs
2. Explanation of the ICs
3. Explanation of the example EALs and their relationship to the ICs
4. Interface between the ICs/EALs and the Off-site Dose Calculation Manual (ODCM)
5. Monitor setpoints versus EAL thresholds.
6. The impact of meteorology
7. The impact of source term

#### 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 off-site 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-established classification thresholds match those that are present at the time of the incident.

Section 3.3 of NEI 07-01 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 07-01 effluent ICs/EALs work together to provide for reasonably accurate and timely emergency classifications. While some aspects of the radiological effluent EALs may appear to be potentially unconservative, one also needs to consider IC/EALs in other recognition categories that compensate for this condition. During site-specific implementation of these ICs/EALs, changes to some of these aspects might appear advantageous. While

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### Basis for Radiological Effluent Initiating Conditions

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 07-01. The IC and the fundamental basis for the ultimate classification for the four classifications are:

General (AG1)	Off-site Dose Resulting from an Actual or IMMEDIATE Release of Gaseous Radioactivity Exceeds 1000 mR TEDE or 5000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.
Site Area (AS1)	Off-site Dose Resulting from an Actual or IMMEDIATE Release of Gaseous Radioactivity Exceeds 100 mR TEDE or 500 mR Thyroid CDE for the Actual or Projected Duration of the Release.
Alert (AA1)	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times Off-site Dose Calculation Manual for 15 Minutes or Longer.
NOUE (AU1)	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times Off-site Dose Calculation Manual for 60 Minutes or Longer.

The fundamental basis of AU1 and AA1 ICs differs from that for AS1 and AG1 ICs. It is important to understand the differences.

- Off-site Dose Calculation Manuals (ODCM) 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 off-site 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 off-site 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 off-site dose rates as **symptoms** that the ODCM limits may be exceeded, the IC, and the classification, are **NOT** concerned with the particular value of off-site 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 off-site 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 Emergency and General Emergency.

#### A.3 Example Emergency Action Level Thresholds

For each of the classifications, NEI 07-01 provides some Example Emergency Action Level Thresholds 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:



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1. The EAL corresponds numerically to the threshold in the respective IC. For example, a field survey result of 1000 mrem/hr for a projected release duration 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.
- 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 off-site consequences of a release.
- Dose Assessment Results: These example EALs are included to provide a means to address classifications based on dose assessments.

#### A.3.1 Effluent Monitor Readings

As noted above, these EALs are pre-calculated values that correspond to the condition identified in the IC for a given set of assumptions. The degree of correlation is dependent on how well the assumed parameters (e.g., meteorology, source term, etc.) represent the actual parameters at the time of the emergency.

##### **AS1 and AG1**

Classifications should be made under these EALs if VALID (e.g., channel check, comparison to redundant/diverse indication, etc.) effluent radiation monitor readings exceed the pre-calculated thresholds. In a change from previous versions of this methodology, confirmation by dose assessments is no longer required as a prerequisite to the classification. 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 serve to confirm the validity of the effluent radiation monitor EAL, may indicate that an escalation to a higher classification is necessary, or may indicate that the classification wasn't warranted. AS1 and AG1 both provide that, if dose assessment results are available, the classification should be based on the basis of the dose assessment result rather than the effluent radiation monitor EAL.

##### **AU1 and AA1**

ODCMs provide a methodology for determining default and batch-specific effluent monitor alarm setpoints pursuant to Standard Technical Specification (STS) 3.3.3.9. These setpoints are intended to show that releases are within Technical Specifications. The applicable limits are 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 X/Q. Since the meteorology data is pre-defined, there is a direct correlation between the monitor setpoints and the ODCM limits. Although the actual X/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 off-site 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

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### **Basis for Radiological Effluent Initiating Conditions**

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 these limits.

To obtain the EAL thresholds, multiply the ODCM setpoint for each monitor by 2 (AU1) or 200 (AA1). It would be preferable to reference "*2 times ODCM Setpoint*" or "*200 times ODCM Setpoint*" as the EAL 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 a "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 EAL thresholds are:

- The "2 times" and "200 times" multiples could be increased to address the reduced setpoints. For example, if the stack monitor were set to 50% of the ODCM limit, the EAL threshold could be set to "4 times" and "400 times" the setpoint on that monitor.
- The reduced setpoints could be ignored and the "2 times" and "200 times" multiples used as specified. While numerically conservative, using a single set of multipliers would probably be desirable from a human engineering standpoint.

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

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 basis of 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 these 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 off-site 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.

#### A.3.2 Perimeter Monitor, Field Survey Results, Dose Projection Results

##### **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. If the dose assessment results are available at the time that the classification is made, the results should be used in conjunction with this EAL for classifying the event rather than the effluent radiation monitor EAL.

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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  $\beta$ - $\gamma$  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.

#### **AU1 and AA1**

As discussed previously, the threshold in these ICs is based on exceeding a multiple of the ODCM 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 limits may be exceeded. The three example EALs are consistent with the fundamental basis of AU1 and AA1, that of a 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:  $500 \div 8766 = 0.057$  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, e.g., an effluent monitor EAL.

#### 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 NUMARC/NESP 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 "2 times ODCM Setpoint" or "200 times 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 EAL 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 ODCM controls would not exceed a monitor EAL threshold.
- To eliminate the possibility of a planned release (e.g., containment / drywell purge) resulting in effluent radiation monitor readings that exceed a classification threshold that was based on a

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different calculation method. ODCMs typically require specific alarm setpoints for such releases. If the release can be authorized under the provisions of the ODCM, an emergency classification is not warranted. If the monitor EAL threshold is indexed to the ODCM setpoint (e.g., "...2 times ODCM setpoint...") the monitor EAL will always change in step with the ODCM setpoint.

#### 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 EAL thresholds will not be implemented as actual alarm setpoints, but would be tabulated in the classification procedure. If the monitor EAL 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 EAL thresholds. The NEI 07-01 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 EAL threshold could be used as the alarm setpoint.

#### A.6 The Impact of Meteorology

The existence of uncertainty between actual event meteorology and the meteorology assumed in establishing the EALs was identified above. It is important to note that uncertainty is present regardless of the meteorology data set assumed. The magnitude of the potential difference and, hence, the degree of conservatism will depend on the data set selected. Data sets that are intended to ensure low probability of under-conservative assessments have a high probability of being over-conservative. For nuclear power plants, there are different sets of meteorological data used for different purposes. The two primary sets are:

- For accident analyses purposes, sector X/Q values are set at that value that is exceeded only 0.5% of the hours wind blows into the sector. The highest of the 16 sector values is the maximum sector X/Q value. The site X/Q value is set at that value that is exceeded only 5% of the hours for all sectors. The higher of the sector or site X/Q values is used in accident analyses.
- For routine release situations, annual average X/Q values are calculated for specified receptor locations and at standard distances in each of the 16 radial sectors. In setting ODCM alarm set points, the annual average X/Q value for the most restrictive receptor at or beyond the site boundary is used. The sector annual average X/Q value is normalized for the percentage of time that the wind blows into that sector. In an actual event, the wind direction may be into the affected sector for the entire release duration. Many sites experience typical sector X/Qs that are 10-20 times higher than the calculated annual average for the sector.

In developing the effluent EALs, the NEI EAL Task Force elected to use annual average meteorology for establishing effluent monitor EAL thresholds. This decision was based on the following considerations.

- Use of the accident X/Qs, may be too conservative. For some sites, the difference between the accident X/Q and the annual average X/Q can be a factor of 100-1000. With this difference in magnitude, the calculated monitor EALs for AS1 or AG1 might actually be less than the ODCM alarm setpoints, resulting in unwarranted classifications for releases that might be in compliance with ODCM limits.
- The ODCM is based in part on annual average X/Q (non-normalized). ODCMs already provide alarm setpoints based on annual average X/Q that could be used for AU1 and AA1.

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- Use of a X/Q more restrictive than the X/Q used to establish ODCM alarm setpoints could create a situation in which the EAL value would be less than the ODCM setpoint. In this case, the operators would have no alarm indication to alert them of the emergency condition.
- Use of one X/Q value for AU1 and AA1 and another for AS1 and AG1 might result in monitor EALs that would not progress from low to high classifications. Instead, the AS1 and AA1 EALs might overlap.

Plant specific consideration must be made to determine if annual average meteorology is adequately conservative for site-specific use. If not one of the two more conservative techniques described above should be selected. It is incumbent upon the licensee to ensure that the selection is properly implemented to provide consistent classification escalation.

The impact of the differences between the assumed annual average meteorology and the actual meteorology depends on the particular EAL.

- For the AU1 and AA1 effluent monitor EALs, there is no impact since the IC and the EALs are based on annual average meteorology by definition.
- For the field survey, perimeter monitor, and dose assessment results EALs in AS1 and AG1, there is no impact since the IC and these EALs are based on actual meteorology.
- For the AS1 and AG1 effluent monitor EALs, there may be differences since the IC is based on actual meteorology and the monitor EALs are calculated on the basis of annual average meteorology or, on a site-specific basis, one of the more conservative derivatives of annual average meteorology. This is considered as acceptable in that dose assessments using actual meteorology will be initiated for significant radioactivity releases. Needed escalations can be based on the results of these assessments. As discussed previously, this delay was deemed to be acceptable since in significant release situations, the plant condition EALs should provide the anticipatory classifications necessary for the implementation of off-site protective measures.
- For the field survey, perimeter monitor, and dose assessment results EALs in AU1 and AA1, there is an impact. These three EALs are dependent on actual meteorology. However, the threshold values for all of the AU1 and AA1 EALs are based on the assumption of annual average meteorology. If the actual and annual average meteorology were equal, the IC and all of the EALs would correlate. Since it is likely that the actual meteorology will exceed the annual average meteorology, there will be numerical inconsistencies between these EALs and the IC. 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.

#### A.7 The Impact of Source Term

The ODCM methodology should be used for establishing the monitor EAL thresholds for these ICs. 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.

For AS1 and AG1, the bases suggests the use of the same source terms used for establishing monitor EAL thresholds for AU1 and AA1, or an accident source term if deemed appropriate. This guidance is provided to promote proper escalations, use realistic values, and correlation between rad monitor values and dose assessment results. This guidance is provided to avoid potential overlaps between effluent monitor EALs for

## **Appendix A**

### Basis for Radiological Effluent Initiating Conditions

AA1 and AS1. Other source terms may be appropriate to achieve these goals. In any case, efforts should be made to obtain and use best estimate (For Example: NUREG 1465), as opposed to conservative, source terms for all four ICs.

Even if the same source term is used for all four ICs, the analyst must consider the impact of overly conservative iodine to noble gas ratios. The AU1 and AA1 IC thresholds are based on external noble gas exposure. The AS1 and AG1 ICs are based on either TEDE or thyroid CDE. TEDE includes a contribution from inhalation exposure (i.e., CEDE) while the thyroid CDE is due solely to inhalation exposure. The inhalation exposure is sensitive to the iodine concentration in the source term. Since AU1 and AA1 are based on noble gases, and AS1 and AG1 are dependent on noble gases and iodine, an over conservative iodine to noble gas ratio could result in AS1 and AG1 monitor EAL thresholds that either overlap or are too close to the AA1 monitor EAL thresholds.

As with meteorology, assessment of source terms has uncertainty. This uncertainty is compensated for by the anticipatory classifications provided by ICs in other recognition categories.