ATTACHMENT 2

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO

LICENSE AMENDMENT REQUEST TSCR-166

UPDATED CLEAN COPY OF THE PROPOSED DAEC EAL SCHEME

125 pages follow

Duane Arnold Energy Center
(DAEC)
Emergency Action Levels
Technical Bases Document

TBD, 2018

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DUANE ARNOLD EMERGENCY ACTION LEVELS
TECHNICAL BASIS DOCUMENT

1 BASIS FOR EMERGENCY ACTION LEVELS

1.1 OPERATING REACTORS

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

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

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

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

1.2 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

Selected guidance in NEI 99-01 is applicable to licensees electing to use their 10 CFR 50 emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone ISFSI. The emergency classification levels applicable to an ISFSI are consistent with the requirements of 10 CFR 50 and the guidance in NUREG 0654/FEMA-REP-1. The initiating conditions germane to a 10 CFR 72.32 emergency plan (as described in NUREG-1567) are subsumed within the classification scheme for a 10 CFR 50.47 emergency plan.

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

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

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

1.3 NRC ORDER EA-12-051

The Fukushima Daiichi accident of March 11, 2012, was the result of a tsunami that exceeded the plant's design basis and flooded the site's emergency electrical power supplies and distribution systems. This caused an extended loss of power that severely compromised the key safety functions of core cooling and containment integrity, and ultimately led to core damage in three reactors. While the loss of power also impaired the spent fuel pool cooling function, sufficient water inventory was maintained in the pools to preclude fuel damage from the loss of cooling.

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

NRC Order EA-12-051 states, in part, "All licensees ... shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred." To this end, all licensees must provide:

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

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

NEI 99-01, Revision 6, includes three EALs that reflect the availability of the enhanced spent fuel pool level instrumentation associated with NRC Order EA-12-051. These EALs are included within ICs RA2, RS2, and RG2.

2 KEY TERMINOLOGY USED IN DAEC EAL SCHEME

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

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

2.1 EMERGENCY CLASSIFICATION LEVEL (ECL)

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

Notification of Unusual Event (NOUE)
Alert
Site Area Emergency (SAE)
General Emergency (GE)

2.1.1 Notification of Unusual Event (NOUE)

Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

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

2.1.2 Alert

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

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

2.1.3 Site Area Emergency

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

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

2.1.4 General Emergency (GE)

Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

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

2.2 INITIATING CONDITION (IC)

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

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

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

2.3 EMERGENCY ACTION LEVEL (EAL)

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

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

2.4 FISSION PRODUCT BARRIER THRESHOLD

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

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

Fuel Clad
Reactor Coolant System (RCS)
Containment

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

3 DESIGN OF THE DAEC EMERGENCY CLASSIFICATION SCHEME

3.1 ASSIGNMENT OF EMERGENCY CLASSIFICATION LEVELS (ECLS)

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

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

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

Assessments of the effects and consequences of different types of events and conditions
DAEC abnormal and emergency operating procedure setpoints and transition criteria
DAEC Technical Specification limits and controls
Offsite Dose Assessment Manual (ODAM) radiological release limits
Review of selected Updated Final Safety Analysis Report (UFSAR) accident analyses
Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs)
NUREG 0654, Appendix 1, *Emergency Action Level Guidelines for Nuclear Power Plants*
Industry Operating Experience
Input from DAEC subject matter experts

The following ECL attributes are used to aid in the development of ICs and Emergency Action Levels (EALs). The attributes may be useful in briefing and training settings (e.g., helping an Emergency Director understand why a particular condition is classified as an Alert).

3.1.1 Notification of Unusual Event (NOUE)

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

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

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

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

3.1.2 Alert

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

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

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

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

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

3.1.3 Site Area Emergency (SAE)

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

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

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

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

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

3.1.4 General Emergency (GE)

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

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

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

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

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

3.1.5 Risk-Informed Insights

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

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

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

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

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

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

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

Symptom-based ICs and EALs are parameters or conditions that are measurable over some range using plant instrumentation (e.g., core temperature, reactor coolant level, radiological effluent, etc.). When one or more of these parameters or conditions are off-normal, reactor operators will implement procedures to identify the probable cause(s) and take corrective action.

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

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

3.3 DAEC-SPECIFIC ORGANIZATION AND PRESENTATION OF GENERIC INFORMATION

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

R - Abnormal Radiation Levels / Radiological Effluent – Section 6
C - Cold Shutdown / Refueling System Malfunction – Section 7
E - Independent Spent Fuel Storage Installation (ISFSI) – Section 8
F - Fission Product Barrier – Section 9
H - Hazards and Other Conditions Affecting Plant Safety – Section 10
S - System Malfunction – Section 11

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

The following information and guidance is provided for each IC:

ECL – the assigned emergency classification level for the IC.

Initiating Condition – provides a summary description of the emergency event or condition.

Operating Mode Applicability – Lists the modes during which the IC and associated EAL(s) are applicable (i.e., are to be used to classify events or conditions).

Emergency Action Level(s) – Provides examples of reports and indications that are considered to meet the intent of the IC.

For Recognition Category F, the fission product barrier thresholds are presented in tables and arranged by fission product barrier and the degree of barrier challenge (i.e., potential loss or loss). This presentation method shows the synergism among the thresholds, and supports accurate assessments.

Basis – Provides background information that explains the intent and application of the IC and EALs. In some cases, the basis also includes relevant source information and references.

3.4 IC AND EAL MODE APPLICABILITY

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

The following table shows which Recognition Categories are applicable in each plant mode. The ICs and EALs for a given Recognition Category are applicable in the indicated modes.

MODE APPLICABILITY MATRIX

	Recognition Category					
Mode	**R**	**C**	**E**	**F**	**H**	**S**
Power Operations	X		X	X	X	X
Startup	X		X	X	X	X
Hot Shutdown	X		X	X	X	X
Cold Shutdown	X	X	X		X	
Refueling	X	X	X		X	
Defueled	X	X	X		X	

DAEC Operating Modes

Power Operations (1): Mode Switch in Run

Startup (2): Mode Switch in Startup/Hot Standby or Refuel (with all vessel head closure bolts fully tensioned)

Hot Shutdown (3): Mode Switch in Shutdown, Average Reactor Coolant Temperature >212 °F (with all vessel head closure bolts fully tensioned)

Cold Shutdown (4): Mode Switch in Shutdown, Average Reactor Coolant Temperature ≤ 212 °F (with all vessel head closure bolts fully tensioned)

Refueling (5): Mode Switch in Shutdown or Refuel (with one or more vessel head closure bolts less than fully tensioned)

4 DEVELOPMENT OF THE DAEC EMERGENCY CLASSIFICATION SCHEME

4.1 GENERAL DEVELOPMENT PROCESS

The DAEC ICs and EALs were developed to be unambiguous and readily assessable.

The IC is the fundamental event or condition requiring a declaration. The EAL(s) is the pre-determined threshold that defines when the IC is met.

Useful acronyms and abbreviations associated with the DAEC emergency classification scheme are presented in Appendix A, Acronyms and Abbreviations.

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

4.2 CRITICAL CHARACTERISTICS

When crafting the scheme, DAEC ensured that certain critical characteristics have been met. These critical characteristics are listed below.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are consistent with industry guidance; while the actual wording may be different, the classification intent is maintained. With respect to Recognition Category F, DAEC includes a user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. The user-aid logic is consistent with the classification logic presented in Section 9.
- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are technically complete and accurate (i.e., they contain the information necessary to make a correct classification).
- EAL statements use objective criteria and observable values.
- ICs, EALs, Operating Mode Applicability and Note statements and formatting consider human factors and are user-friendly.
- The scheme facilitates upgrading and downgrading of the emergency classification where necessary.
- The scheme facilitates classification of multiple concurrent events or conditions.

4.3 INSTRUMENTATION USED FOR EALS

DAEC incorporated instrumentation that is reliable and routinely maintained in accordance with site programs and procedures. Alarms referenced in EAL statements are those that are the most operationally significant for the described event or condition.

EAL setpoints are within the calibrated range of the referenced instrumentation, and consider any automatic instrumentation functions that may impact accurate EAL assessment. In addition, EAL setpoint values do not use terms such as "off-scale low" or "off-scale high" since that type of reading may not be readily differentiated from an instrument failure.

4.4 EAL/THRESHOLD REFERENCES TO AOP AND EOP SETPOINTS/CRITERIA

Some of the criteria/values used in several EALs and fission product barrier thresholds are drawn from DAEC AOPs and EOPs. This approach is intended to maintain good alignment between operational diagnoses and emergency classification assessments. Appropriate administrative controls are in place to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required.

5 GUIDANCE ON USING THE DAEC EALS

5.1 GENERAL CONSIDERATIONS

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

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

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

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

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

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

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

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

5.2 CLASSIFICATION METHODOLOGY

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

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

5.3 CLASSIFICATION OF MULTIPLE EVENTS AND CONDITIONS

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. Additionally, there is no "additive" effect from multiple EALs meeting the same ECL. For example:

> If two Alert EALs are met, an Alert should be declared.

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

5.4 CONSIDERATION OF MODE CHANGES DURING CLASSIFICATION

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

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

5.5 CLASSIFICATION OF IMMINENT CONDITIONS

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

5.6 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND DOWNGRADING

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

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

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

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

5.7 CLASSIFICATION OF SHORT-LIVED EVENTS

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

5.8 CLASSIFICATION OF TRANSIENT CONDITIONS

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

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

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

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

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

5.9 AFTER-THE-FACT DISCOVERY OF AN EMERGENCY EVENT OR CONDITION

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

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

5.10 RETRACTION OF AN EMERGENCY DECLARATION

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

6 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

RU1

ECL: Notification of Unusual Event

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the ODAM limits for 60 minutes or longer.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

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

RU1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "NOUE" for 60 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RU1.2 Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.3 Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODAM limits for 60 minutes or longer.

Definitions:

None

Basis:

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

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

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

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

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

EAL RU1.1 - This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

EAL RU1.2 - This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL will typically be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

EAL RU1.3 - This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analysis or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC RA1.

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel.

Operating Mode Applicability: All

Emergency Action Levels:

RU2.1 a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:

- Report to control room (visual observation)
- Fuel pool level indication (LI-3413) less than 36 feet and lowering
- WR GEMAC Floodup indication (LI-4541) coming on scale

AND

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

- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164
- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool and fuel transfer canal.

Basis:

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

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

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

During preparation for reactor cavity flood up prior to entry into refuel mode, reactor vessel level instrument LI-4541 (WR GEMAC, FLOODUP) on control room panel 1C04 is placed in service by I&C personnel connecting a compensating air signal after the reference leg is disconnected from the reactor head. Normal refuel water level is above the top of the span of this flood up level indicator. A valid indication (e.g., not due to loss of compensating air signal or other instrument channel failure) of reactor cavity level coming on span for this instrument is used at DAEC as an indicator of uncontrolled reactor cavity level decrease.

DAEC Technical Specifications require a minimum of 36 feet of water in the spent fuel pool when moving irradiated fuel into the secondary containment. During refueling, the gates between the reactor cavity and the refueling cavity are removed and the spent fuel pool level indicator LI- 3413 is used to monitor refueling water level. Procedures require that a normal refueling water level be maintained at 37 feet 5 inches. A low level alarm actuates when spent fuel pool level drops below 37 feet 1 inch. Symptoms of inventory loss at DAEC include visual observation of decreasing water levels in reactor cavity or spent fuel storage pool, Reactor Building (RB) fuel storage pool radiation monitor or refueling area radiation monitor alarms, observation of a decreasing trend on the spent fuel pool water level indicator, and actuation of the spent fuel pool low water level alarm. To eliminate minor level perturbations from concern, DAEC uses LI-3413 indicated water level below 36 feet and lowering.

Increased radiation levels can be detected by the local area radiation monitors surrounding the spent fuel pool and refueling cavity areas. Applicable area radiation monitors are those listed in AOP 981.

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

Escalation of the emergency classification level would be via IC RA2.

RA1

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RA1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "Alert" for 15 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RA1.2 Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond SITE BOUNDARY. [Preferred]

RA1.3 Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.

RA1.4 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

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

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

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

This IC is modified by a note that EAL RA1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions.

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

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

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

Escalation of the emergency classification level would be via IC RS1.

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels:

RA2.1 Uncovery of irradiated fuel in the REFUELING PATHWAY.

RA2.2 Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by Hi Rad alarm for **ANY** of the following ARMs:

- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164

OR

Reading greater than 5 R/hr on **ANY** of the following radiation monitors (in Mode 5 only):

- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

RA2.3 Lowering of spent fuel pool level to 25.17 feet.

Definitions:

REFUELING PATHWAY – The reactor refueling cavity, spent fuel pool and fuel transfer canal.

Basis:

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

Expected radiation monitor alarm(s) during preplanned transfer of highly radioactive material through the affected areas are not considered valid alarms for the purpose of comparison to these EALs.

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

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

EAL RA2.1

This EAL escalates from RU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used. Classification of an event using this EAL should be based on the totality of available indications, reports, and observations.

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

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

EAL RA2.2

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

Threshold values for the Drywell monitors are only applicable in Mode 5 since the calculated radiation levels from damage to irradiated fuel would be masked by the typical background levels on these monitors during plant operation, and mechanical damage to a fuel assembly in the vessel can only happen with the reactor head removed.

EAL RA2.3

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

Escalation of the emergency classification level would be via ICs RS1 or RS2.

ECL: Alert

Initiating Condition: Radiation levels that impede access to areas necessary for normal plant operation.

Operating Mode Applicability: All

Emergency Action Levels:

RA3.1 Dose rate greater than 15 mR/hr in **ANY** of the following areas:

- Control Room (RM-9162)
- Central Alarm Station (by survey)

Definitions:

None

Basis:

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

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

ECL: Site Area Emergency

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

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RS1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RS1.2 Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]

RS1.3 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

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

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

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

This IC is modified by a note that EAL RS1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions. However, if Kaman monitor readings are sustained for 15 minutes or longer and the required MIDAS dose assessments cannot be completed within this period, then the declaration can be made using Kaman readings PROVIDED the readings are not from an isolated flow path.

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

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

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. If Kaman readings are not valid, field survey results may be utilized to assess this IC using EAL RS1.3.

Escalation of the emergency classification level would be via IC RG1.

ECL: Site Area Emergency

Initiating Condition: Spent fuel pool level at 16.36 feet.

Operating Mode Applicability: All

Emergency Action Levels:

RS2.1 Lowering of spent fuel pool level to 16.36 feet.

Definitions:

None

Basis:

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

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

Escalation of the emergency classification level would be via IC RG1 or RG2.

RG1

ECL: General Emergency

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

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RG1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RG1.2 Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]

RG1.3 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

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

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

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

This IC is modified by a note that EAL RG1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions. However, if Kaman monitor readings are sustained for 15 minutes or longer and the required MIDAS dose assessments cannot be completed within this period, then the declaration can be made using Kaman readings PROVIDED the readings are not from an isolated flow path.

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

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

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. If Kaman readings are not valid, field survey results may be utilized to assess this IC using EAL RG1.3.

ECL: General Emergency

Initiating Condition: Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.

Operating Mode Applicability: All

Emergency Action Levels:

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

RG2.1 Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.

Definitions:

None

Basis:

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

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

7 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of RPV inventory for 15 minutes or longer.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

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

CU1.1 UNPLANNED loss of reactor coolant results in RPV level less than a required lower limit for 15 minutes or longer.

CU1.2 a. RPV level cannot be monitored.

AND

b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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

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

EAL CU1.1 recognizes that the minimum required RPV level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

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

EAL CU1.2 addresses a condition where all means to determine RPV level have been lost. If all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RSC inventory loss was occurring by observing sump and Suppression Pool level changes. The drywell floor and equipment drain sumps, reactor building equipment and floor drain sumps receive all liquid waste from floor and equipment drains inside the primary containment and reactor building. A rise in Suppression Pool water level may be indicative of valve misalignment or leakage in systems that discharge to the Torus. Sump and Suppression Pool 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.

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

CU2

ECL: Notification of Unusual Event

Initiating Condition: Loss of all but one AC power source to essential buses for 15 minutes or longer.

Operating Mode Applicability: 4, 5, Defueled

Emergency Action Levels:

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

CU2.1 a. AC power capability to 1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.

AND

b. Any additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

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

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

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED increase in RCS temperature.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

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

CU3.1 UNPLANNED increase in RCS temperature to greater than 212°F.

CU3.2 Loss of **ALL** RCS temperature and RPV level indication for 15 minutes or longer.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

Basis:

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or the inability to determine RCS temperature and level, represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

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

EAL CU3.1 involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

EAL CU3.2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

ECL: Notification of Unusual Event

Initiating Condition: Loss of Vital DC power for 15 minutes or longer.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

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

CU4.1 Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, “required” means the Vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category R.

ECL: Notification of Unusual Event

Initiating Condition: Loss of all onsite or offsite communications capabilities.

Operating Mode Applicability: 4, 5, Defueled

Emergency Action Levels:

CU5.1 Loss of **ALL** of the following onsite communication methods:

- Plant Operations Radio System
- In-Plant Phone System
- Plant Paging System (Gaitronics)

CU5.2 Loss of **ALL** of the following offsite response organization communications methods:

- DAEC All-Call phone
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- FTS Phone system

CU5.3 Loss of **ALL** of the following NRC communications methods:

- FTS Phone system
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system

Basis:

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

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

EAL CU5.1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL CU5.2 addresses a total loss of the communications methods used to notify all offsite response organizations of an emergency declaration. The offsite response organizations referred to here are the State of Iowa, Linn County, and Benton County.

EAL CU5.3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

ECL: Alert

Initiating Condition: Loss of RPV inventory.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

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

CA1.1 Loss of RPV inventory as indicated by level less than 119.5 inches.

CA1.2 a. RPV level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool due to a loss of RPV inventory.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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

For EAL CA1.1, a lowering of water level below 119.5 inches indicates that operator actions have not been successful in restoring and maintaining RPV water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, EAL CA1.1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

For EAL CA1.2, the inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, the operators would need to determine that RCS inventory loss was occurring by observing sump and Suppression Pool level changes. The drywell floor and equipment drain sumps, reactor building equipment and floor drain sumps receive all liquid waste from floor and equipment drains inside the primary containment and reactor building. A rise in Suppression Pool water level may be indicative of valve misalignment or leakage in systems that discharge to the Torus. Sump and Suppression Pool level increases must be

evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1

If the RPV inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

ECL: Alert

Initiating Condition: Loss of all offsite and all onsite AC power to essential buses for 15 minutes or longer.

Operating Mode Applicability: 4, 5, Defueled

Emergency Action Levels:

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

CA2.1 Loss of **ALL** offsite and **ALL** onsite AC Power to 1A3 and 1A4 buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the emergency classification level would be via IC CS1 or RS1.

ECL: Alert

Initiating Condition: Inability to maintain the plant in cold shutdown.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

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

CA3.1 UNPLANNED increase in RCS temperature to greater than 212°F for greater than the duration specified in Table C-2.

Table C-2 RCS Heat-up Duration Thresholds

RCS Integrity	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact	Not applicable	60 minutes*
Not intact	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.

CA3.2 UNPLANNED RCS pressure increase greater than 10 psig due to a loss of RCS cooling.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

Basis:

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

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

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

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

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because

1) the evaporated reactor coolant may be released directly into the Containment atmosphere and subsequently to the environment, and
2) there is reduced reactor coolant inventory above the top of irradiated fuel.

EAL CA3.2 provides a pressure-based indication of RCS heat-up.

Escalation of the emergency classification level would be via IC CS1 or RS1.

CA6

ECL: Alert

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

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

CA6.1 a. The occurrence of **ANY** of the Table C-3 hazardous events:

Table C-3 Hazardous Events

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

AND

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

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode,

OR

- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Definitions:

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

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction, or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria CA6.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under CA6 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC RS1.

ECL: Site Area Emergency

Initiating Condition: Loss of RPV inventory affecting core decay heat removal capability.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

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

CS1.1 a. CONTAINMENT CLOSURE not established.

AND

b. RPV level less than +64 inches

CS1.2 a. CONTAINMENT CLOSURE established.

AND

b. RPV level less than +15 inches

CS1.3 a. RPV level cannot be monitored for 30 minutes or longer.

AND

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

- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery

Definitions:

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified reactor vessel levels of EALs CS1.1.b and CS1.2.b reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

In the Cold Shutdown and Refueling Modes, LT/LI-4559, 4560, and 4561 (RX VESSEL NARROW RANGE LEVEL) instruments read up to 22" high due to hot calibrations. LI-4541 (WR GEMAC, FLOODUP) should be used in these Modes for comparison to EAL thresholds since it is calibrated cold and reads accurately. If normal means of RPV level indication are not available due to plant evolutions, redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

In EAL CS1.3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RPV.

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

Escalation of the emergency classification level would be via IC CG1 or RG1.

ECL: General Emergency

Initiating Condition: Loss of RPV inventory affecting fuel clad integrity with containment challenged.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

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

CG1.1 a. RPV level less than +15 inches for 30 minutes or longer.

AND

b. **ANY** indication from the Secondary Containment Challenge Table C-1.

CG1.2 a. RPV level cannot be monitored for 30 minutes or longer.

AND

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

- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr.
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery.

AND

c. **ANY** indication from the Secondary Containment Challenge Table C-1.

Table C-1 Secondary Containment Challenge
• CONTAINMENT CLOSURE not established* • Drywell Hydrogen or Torus Hydrogen greater than 6% **AND** Drywell Oxygen or Torus Oxygen greater than 5% • UNPLANNED increase in containment pressure • Secondary containment radiation monitors above max safe operating limits (MSOL) of EOP 3, Table 6

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

Definitions:

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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

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

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

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

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

In EAL CG1.2.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

For EAL CG1.2.b, the calculated radiation level on the Drywell Monitors (9184A/B) is without the reactor head in place. Calculated in radiation levels with the reactor head in place are below the normal variation in background readings of these monitors.

The inability to monitor RPV level may be caused by instrumentation and/or power failures or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RPV.

For the Containment Challenge Table, Secondary Containment max safe operating (MSOL) limits from EOP 3 are defined as the highest parameter value at which neither: (1) equipment necessary for the safe shutdown of the plant will fail nor (2) personnel access necessary for the safe shutdown of the plant will be precluded.

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

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

ECL: Notification of Unusual Event

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: All

Emergency Action Levels:

E-HU1.1 Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by a radiation reading greater than the values shown on Table E-1 on the spent fuel cask.

Table E-1 Cask Dose Rates	
61BT DSC	
3 feet from HSM Surface	800 mrem/hr
Outside HSM Door – Centerline of DSC	200 mrem/hr
End Shield Wall Exterior	40 mrem/hr

Definition:

CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

Basis:

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

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

9 FISSION PRODUCT BARRIER ICS/EALS

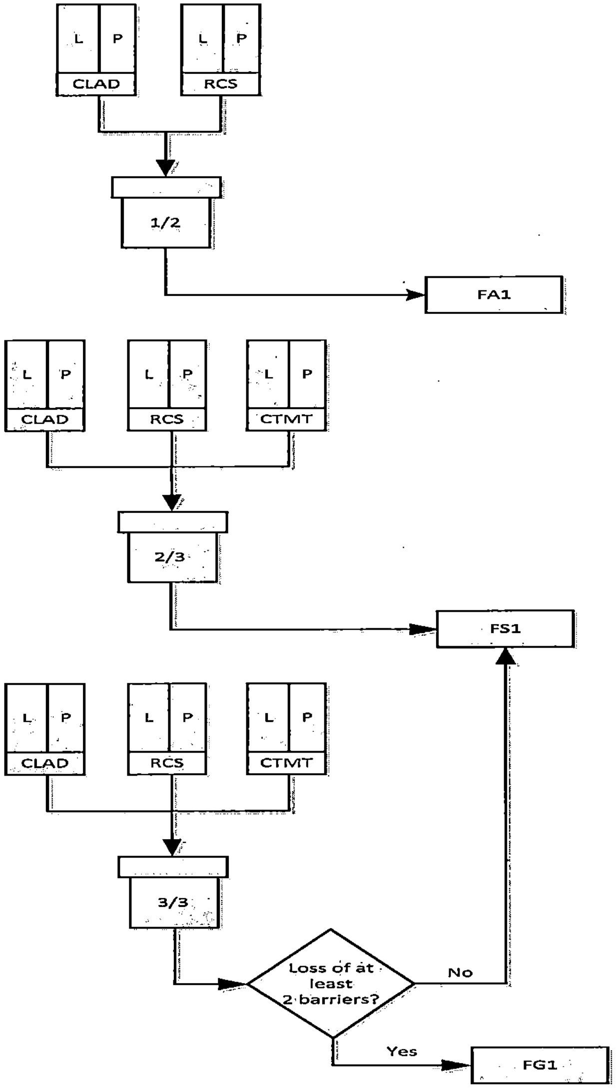

Table F-1: DAEC EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

FA1 ALERT	FS1 SITE AREA EMERGENCY	FG1 GENERAL EMERGENCY
ANY Loss **OR ANY** Potential Loss of **EITHER** the Fuel Clad **OR** RCS barrier.	Loss **OR** Potential Loss of **ANY** two barriers.	Loss of **ANY** two barriers **AND** Loss **OR** Potential Loss of the third barrier
Operating Mode Applicability: 1, 2, 3	**Operating Mode Applicability:** 1, 2, 3	**Operating Mode Applicability:** 1, 2, 3

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. RCS Activity		**1. Primary Containment Conditions**		**1. Primary Containment Conditions**	
A. Coolant activity greater than 300 µCi/gm dose equivalent I-131.	Not Applicable	A. Primary containment pressure greater than 2 psig due to RCS leakage.	Not Applicable	A. UNPLANNED rapid drop in Drywell pressure following Drywell pressure rise **OR** B. Drywell pressure response not consistent with LOCA conditions. **OR** C. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal **OR** D. Intentional primary containment venting per EOPs	A. Torus pressure greater than 53 psig **OR** B. Drywell or Torus H2 cannot be determined to be less than 6% and Drywell **OR** Torus O2 cannot be determined to be less than 5% **OR** C. HCL (Graph 4 of EOP 2) exceeded.

Fuel Clad Barrier		**RCS Barrier**		**Containment Barrier**	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. RPV Water Level		**2. RPV Water Level**		**2. RPV Water Level**	
A. SAG entry is required	A. RPV water level cannot be restored and maintained above +15 inches **OR** cannot be determined.	A. RPV water level cannot be restored and maintained above +15 inches **OR** cannot be determined.	Not Applicable	Not Applicable	A. SAG entry is required
3. Not Applicable		**3. RCS Leak Rate**		**3. Primary Containment Isolation Failure**	
Not Applicable	Not Applicable	A. UNISOLABLE break in Main Steam, HPCI, Feedwater, RWCU, or RCIC as indicated by the failure of both isolation valves in **ANY** one line to close **AND EITHER**: • High MSL flow or steam tunnel temperature annunciators **OR** • Direct report of steam release **OR** B. Emergency RPV Depressurization required.	A. UNISOLABLE primary system leakage that results in exceeding the Max Normal Operating Limit (MNOL) of EOP 3, Table 6 for **EITHER** of the following: • Temperature **OR** • Radiation Level	A. UNISOLABLE primary system leakage that results in exceeding the Max Safe Operating Limit (MSOL) of EOP 3, Table 6 for **EITHER** of the following: • Temperature **OR** • Radiation Level	Not Applicable

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
4. Primary Containment Radiation		**4. Primary Containment Radiation**		**4. Primary Containment Radiation**	
A. Drywell Monitor (9184A/B) reading greater than 2000 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 200 R/hr	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5 R/hr after reactor shutdown	Not Applicable	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5000 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 500 R/hr
5. Other Indications		**5. Other Indications**		**5. Other Indications**	
A. Fuel damage assessment indicates at least 5% fuel clad damage.	Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Fuel damage assessment indicates at least 20% fuel clad damage.
6. Emergency Director Judgment		**6. Emergency Director Judgment**		**6. Emergency Director Judgment**	
A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis Information For
DAEC EAL Fission Product Barrier Table F-1

DAEC FUEL CLAD BARRIER THRESHOLDS:

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

1. RCS Activity

Loss 1.A

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

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

There is no Potential Loss threshold associated with RCS Activity.

2. RPV Water Level

Loss 2.A

The Loss threshold represents any EOP requirement for entry into the Severe Accident Guidelines.

This is identified in the BWROG EPGs/SAGs when adequate core cooling cannot be assured.

Potential Loss 2.A

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

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

DAEC FUEL CLAD BARRIER THRESHOLDS (cont.):

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

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

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA6 or SS6 will dictate the need for emergency classification.

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

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

DAEC FUEL CLAD BARRIER THRESHOLDS (cont.):

4. Primary Containment Radiation

Loss 4.A and Loss 4.B

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

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

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

5. Other Indications

Loss 5.A

Results obtained from procedure PASAP 7.2, Fuel Damage Assessment, indicate at least 5% fuel clad damage.

There is no Potential Loss threshold associated with Other Indications.

6. Emergency Director Judgment

Loss 6.A

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

Potential Loss 6.A

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

DAEC RCS BARRIER THRESHOLDS:

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

1. Primary Containment Conditions

Loss 1.A

2 psig is the drywell high pressure scram setpoint which indicates a LOCA by automatically initiating ECCS.

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

2. RPV Water Level

Loss 2.A

+15 inches corresponds to the top of active fuel (TAF) and is used in the EOPs to indicate challenge to core cooling.

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

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

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

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

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

3. RCS Leak Rate

Loss Threshold 3.A

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

Loss Threshold 3.B

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

Potential Loss Threshold 3.A

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

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

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

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

DAEC RCS BARRIER THRESHOLDS (cont.):

4. Primary Containment Radiation

Loss 4.A

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

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

5. Other Indications

There are no Loss or Potential Loss thresholds associated with Other Indications.

6. Emergency Director Judgment

Loss 6.A

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

Potential Loss 6.A

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

DAEC CONTAINMENT BARRIER THRESHOLDS:

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

1. Primary Containment Conditions

Loss 1.A and 1.B

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

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

Loss 1.C

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).

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

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

Loss 1.D

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

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

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):

Potential Loss 1.A

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

Potential Loss 1.B

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

Potential Loss 1.C

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

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,
  OR
- Suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

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

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):

2. RPV Water Level

There is no Loss threshold associated with RPV Water Level.

Potential Loss 2.A

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold 2.A. The Potential Loss requirement for Primary Containment Flooding indicates adequate core cooling cannot be restored and maintained and that core damage is possible. BWR EPGs/SAGs specify the conditions that require primary containment flooding. When primary containment flooding is required, the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to restore and maintain adequate core cooling.

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

3. Primary Containment Isolation Failure

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

Loss 3.A

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

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

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

There is no Potential Loss threshold associated with RCS Leak Rate.

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):

4. Primary Containment Radiation

There is no Loss threshold associated with Primary Containment Radiation.

Potential Loss 4.A

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

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

5. Other Indications

There is no Loss threshold associated with Other Indications

Potential Loss 5.A

Results obtained from procedure PASAP 7.2, Fuel Damage Assessment, indicate at least 25% fuel clad damage. PASAP 7.2 only shows whether fuel damage is greater than or less than 25%, thus this indication is not likely to be declared before containment barrier potential loss 4.A which indicates 20% fuel damage. However, this potential loss threshold adds an additional layer of diversity to the scheme.

6. Emergency Director Judgment

Loss 6.A

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

Potential Loss 6.A

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

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10 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

HU1

ECL: Notification of Unusual Event

Initiating Condition: Confirmed SECURITY CONDITION or threat.

Operating Mode Applicability: All

Emergency Action Levels:

HU1.1 A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by DAEC Security Shift Supervision.

HU1.2 Notification of a credible security threat directed at DAEC.

HU1.3 A validated notification from the NRC providing information of an aircraft threat.

Definitions:

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.

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

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

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

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

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

EAL HU1.1 references DAEC Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.390 information.

EAL HU1.2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with Abnormal Operating Procedure (AOP) 914, Security Events.

EAL HU1.3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with Abnormal Operating Procedure (AOP) 914, Security Events.

Emergency plans and implementing procedures are public documents; therefore, EALs do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information is contained in the Security Plan.

Escalation of the emergency classification level would be via IC HA1.

ECL: Notification of Unusual Event

Initiating Condition: Seismic event greater than OBE levels.

Operating Mode Applicability: All

Emergency Action Levels:

HU2.1 Seismic event greater than Operating Basis Earthquake (OBE) as indicated by receipt of the Amber Operating Basis Earthquake Light and the wailing seismic alarm on 1C35.

Definitions:

DESIGN BASIS EARTHQUAKE (DBE): A DBE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

OPERATING BASIS EARTHQUAKE (OBE): An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Design Basis Earthquake (DBE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event. The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

OBE events are detected in accordance with AOP 901. The OBE is associated with a peak horizontal acceleration of ± 0.06g.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

ECL: Notification of Unusual Event

Initiating Condition: Hazardous events

Operating Mode Applicability: All

Emergency Action Levels:

Note: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

HU3.1 A tornado strike within the PROTECTED AREA.

HU3.2 Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

HU3.3 Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

HU3.4 A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

Definitions:

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the Protected Area.

EAL HU3.2 addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

EAL HU3.3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

EAL HU3.4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

ECL: Notification of Unusual Event

Initiating Condition: FIRE potentially degrading the level of safety of the plant.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

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

HU4.1 a. A FIRE is **NOT** extinguished within 15-minutes of **ANY** of the following FIRE detection indications:
  - Report from the field (i.e., visual observation)
  - Receipt of multiple (more than 1) fire alarms or indications
  - Field verification of a single fire alarm

AND

b. The FIRE is located within **ANY** Table H-1 plant rooms or areas

HU4.2 a. Receipt of a single fire alarm with no other indications of a FIRE.

AND

b. The FIRE is located within **ANY** Table H-1 plant rooms or areas

AND

c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

HU4.3 A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

HU4.4 A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Table H-1 Fire Areas

- 1G31 DG and Day Tank Rooms
- 1G21 DG and Day Tank Rooms
- Battery Rooms
- Essential Switchgear Rooms
- Cable Spreading Room
- Torus Room
- Intake Structure
- Pumphouse
- Drywell
- Torus
- NE, NW, SE Corner Rooms
- HPCI Room
- RCIC Room
- RHR Valve Room
- North CRD Area
- South CRD Area
- CSTs
- Control Building
- Remote Shutdown Panel 1C388 Area
- Panel 1C55/56 Area
- SBGT Room

Definitions:

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

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL HU4.1

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

EAL HU4.2

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then EAL HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15 minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

EAL HU4.3

In addition to a FIRE addressed by EAL HU4.1 or EAL HU4.2, a FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.

EAL HU4.4

If a FIRE within the plant or ISFSI PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Basis-Related Requirements from Appendix R and NFPA-805

> Criterion 3 of Appendix A to 10 CFR 50 states in part that "structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> The Nuclear Safety Goal ("NSG") in NFPA 805, Section 1.3.1 states, "The nuclear safety goal is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance because a safe shutdown success path, free of fire damage, must be available to meet the nuclear safety goals, objectives and performance criteria for a fire under any plant operational mode or configuration.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). Even though DAEC has adopted the alternate approach provided by NFPA-805 in lieu of the deterministic requirements of Appendix R, the 30-minutes to verify a single alarm as used in EAL HU4.2 is considered a reasonable amount of time to determine if an actual FIRE exists without presenting a challenge to the nuclear safety performance criteria.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

ECL: Notification of Unusual Event

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

Operating Mode Applicability: All

Emergency Action Levels:

HU6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

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

ECL: Alert

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.

Operating Mode Applicability: All

Emergency Action Levels:

HA1.1 A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the DAEC Security Shift Supervision.

HA1.2 A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Definitions:

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

OWNER CONTROLLED AREA: The site property owned by or otherwise under the control of the licensee.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

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

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

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

The Alert declaration will also heighten the awareness of Offsite Response Organizations, allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72.

EAL HA1.1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against the ISFSI which is located outside the plant PROTECTED AREA.

EAL HA1.2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and offsite response organizations are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with Abnormal Operating Procedure (AOP) 914, Security Events.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information is contained in the Security Plan.

Escalation of the emergency classification level would be via IC HS1.

ECL: Alert

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.

Operating Mode Applicability: All

Emergency Action Level:

HA5.1 An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).

Definitions:

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS5.

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Level:

HA6.1 Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Definitions:

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

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

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

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

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Levels:

HS1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.

Definitions:

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

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

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.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

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

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

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize offsite response organization resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to a HOSTILE ACTION directed at the ISFSI PROTECTED AREA which is located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information is contained in the Security Plan.

Escalation of the emergency classification level would be via IC HG1.

ECL: Site Area Emergency

Initiating Condition: Inability to control a key safety function from outside the Control Room.

Operating Mode Applicability: All

Emergency Action Levels:

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

HS5.1 a. An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).

AND

b. Control of **ANY** of the following key safety functions is not reestablished within 20 minutes.

- Reactivity control
- RPV water level
- RCS heat removal

Definitions:

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the Remote Shutdown Panel (1C388) is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within 20 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

AOP 915, "Shutdown Outside Control Room" provides the following CAUTION – "*For Control Room evacuation as the result of a fire, transfer of control at panels 1C388, 1C389, 1C390, 1C391, and 1C392 is required to be completed within 20 minutes.*"

Escalation of the emergency classification level would be via IC FG1 or CG1.

ECL: Site Area Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.

Operating Mode Applicability: All

Emergency Action Level:

HS6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Definitions:

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

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

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

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

ECL: General Emergency

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility.

Operating Mode Applicability: All

Emergency Action Level:

HG1.1 a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.

AND

b. **EITHER** of the following has occurred:

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

    - Reactivity control
    - RPV water level
    - RCS heat removal

    OR

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

Definitions:

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

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

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

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

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

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

Emergency plans and implementing procedures are public documents; therefore, EALs do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information is contained in the Security Plan.

ECL: General Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.

Operating Mode Applicability: All

Emergency Action Level:

HG6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Definitions:

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

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

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

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

11 SYSTEM MALFUNCTION ICS/EALS

ECL: Notification of Unusual Event

Initiating Condition: Loss of ALL offsite AC power capability to essential buses for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

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

SU1.1 Loss of **ALL** offsite AC power capability to 1A3 **AND** 1A4 buses for 15 minutes or longer.

Definitions:

None

Basis:

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC essential buses. This condition represents a potential reduction in the level of safety of the plant.

The intent of this EAL is to declare a Notification of Unusual Event when offsite power has been lost and both of the emergency diesel generators have successfully started and energized their respective 4kv essential bus.

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

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

Escalation of the emergency classification level would be via IC SA1.

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

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

SU3.1 An UNPLANNED event results in the inability to monitor one or more of the Table S-1 parameters from within the Control Room for 15 minutes or longer.

Table S-1 Safety System Parameters

- Reactor power
- RPV Water Level
- RPV Pressure
- Primary Containment Pressure
- Suppression Pool Level
- Suppression Pool Temperature

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC SA3.

ECL: Notification of Unusual Event

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.

Operating Mode Applicability: 1, 2, 3

Emergency Action Levels:

SU4.1 Pretreatment Offgas System (RM-4104) Hi-Hi Radiation Alarm.

SU4.2 Sample analysis indicates that reactor coolant specific activity is greater than 2.0 µCi/gm dose equivalent I-131 for 12 hours or longer.

Definitions:

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

For EAL SU4.1, RM-4104 Hi-Hi Radiation Alarm has been chosen because it is operationally significant, is readily recognizable by the Control Room Operations Staff, and is set at a level corresponding to noble gas release rate, after 30-minute delay and decay of 1 Ci/sec.

For EAL SU4.2, coolant samples exceeding the 2.0 µCi/gm dose equivalent I-131concentration require prompt action by DAEC Technical Specifications and are representative of minor fuel cladding degradation.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

ECL: Notification of Unusual Event

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Levels:

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

SU5.1 RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.

SU5.2 RCS identified leakage greater than 25 gpm for 15 minutes or longer.

SU5.3 Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.

Definitions:

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

EAL SU5.1 and EAL SU5.2 are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications).

EAL SU5.3 addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These EALs thus apply to leakage into the containment, a secondary-side system) or a location outside of containment.

The leak rate values for each EAL were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). EAL SU5.1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

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

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

ECL: Notification of Unusual Event

Initiating Condition: Automatic or manual scram fails to shutdown the reactor.

Operating Mode Applicability: 1, 2

Emergency Action Levels:

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

SU6.1 a. An automatic scram did not shutdown the reactor.

AND

b. ANY of the following manual actions taken at 1C05 are successful in lowering reactor power below 5% power

- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)

SU6.2 a. A manual scram did not shutdown the reactor.

AND

b. **EITHER** of the following:

1. ANY of the following subsequent manual actions taken at 1C05 are successful in lowering reactor power below 5% power
    - Manual Scram Pushbuttons
    - Mode Switch to Shutdown
    - Alternate Rod Insertion (ARI)

OR

2. A subsequent automatic scram is successful in shutting down the reactor.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control console to shutdown the reactor (e.g., initiate a manual reactor scram quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control console to shutdown the reactor (e.g., initiate a manual reactor scram using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power).

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

ECL: Notification of Unusual Event

Initiating Condition: Loss of ALL onsite or offsite communications capabilities.

Operating Mode Applicability: 1, 2, 3 .

Emergency Action Levels:

SU7.1 Loss of **ALL** of the following onsite communication methods:

- Plant Operations Radio System
- In-Plant Phone System
- Plant Paging System (Gaitronics)

SU7.2 Loss of **ALL** of the following offsite response organization communications methods:

- DAEC All-Call phone
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- FTS Phone system

SU7.3 Loss of **ALL** of the following NRC communications methods:

- FTS Phone system
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system

Basis:

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

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

EAL SU7.1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL SU7.2 addresses a total loss of the communications methods used to notify all offsite response organizations of an emergency declaration. The offsite response organizations referred to here are the State of Iowa, Linn County, and Benton County.

EAL SU7.3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

ECL: Alert

Initiating Condition: Loss of ALL but one AC power source to essential buses for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

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

SA1.1 a. AC power capability to 1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.

AND

b. **ANY** additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

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

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

Escalation of the emergency classification level would be via IC SS1.

SA3

ECL: Alert

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

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

SA3.1 a. An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for 15 minutes or longer.

Table S-1 Safety System Parameters

- Reactor power
- RPV Water Level
- RPV Pressure
- Primary Containment Pressure
- Suppression Pool Level
- Suppression Pool Temperature

AND

b. **ANY** of the Table S-2 transient events are in progress.

Table S-2 Significant Transients

- Automatic or manual runback greater than 25% thermal reactor power
- Electrical load rejection greater than 25% full electrical load
- Reactor scram
- ECCS actuation
- Thermal power oscillations greater than 10%

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or IC RS1.

ECL: Alert

Initiating Condition: Automatic or manual scram fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.

Operating Mode Applicability: 1, 2

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Emergency Action Level:

SA6.1 a. An automatic or manual scram did not shutdown the reactor.

AND

b. ALL of the following manual actions taken at 1C05 are not successful in lowering reactor power below 5% power

- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram. This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (e.g., locally opening breakers). Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles."

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shutdown the reactor is prolonged enough to cause a challenge to the RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power).

SA8

ECL: Alert

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

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of the SAFETY SYSTEM, then this emergency classification is not warranted.

SA8.1 a. The occurrence of **ANY** of the Table S-3 hazardous events:

Table S-3 Hazardous Events

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

AND

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

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode,

OR

- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Definitions:

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

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

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria SA8.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under SA8 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC FS1 or RS1.

ECL: Site Area Emergency

Initiating Condition: Loss of ALL offsite and ALL onsite AC power to essential buses for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

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

SS1.1 Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

ECL: Site Area Emergency

Initiating Condition: Loss of ALL Vital DC power for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

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

SS2.1 Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG2.

SS6

ECL: Site Area Emergency

Initiating Condition: Inability to shutdown the reactor causing a challenge to RPV water level or RCS heat removal.

Operating Mode Applicability: 1, 2

Emergency Action Levels:

SS6.1 a. An automatic or manual scram did not shutdown the reactor.

AND

b. **ALL** of the following manual actions taken at 1C05 are not successful in lowering reactor power below 5% power:

- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)

AND

c. **EITHER** of the following conditions exist:

- RPV level cannot be restored and maintained above -25 inches.

OR

- HCL (Graph 4 of EOP 2) exceeded.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shutdown the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power).

Escalation of the emergency classification level would be via IC RG1 or FG1.

ECL: General Emergency

Initiating Condition: Prolonged loss of ALL offsite and ALL onsite AC power to essential buses.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

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

SG1.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses.

AND

b. **EITHER** of the following:

- Restoration of at least one AC essential bus in less than 4 hours is not likely.

OR

- RPV level cannot be restored and maintained above -25 inches.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a prolonged loss of all power sources to AC essential buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC essential bus by the end of the 4 hour station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one essential bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

ECL: General Emergency

Initiating Condition: Loss of ALL AC and Vital DC power sources for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

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

SG2.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses for 15 minutes or longer.

AND

b. Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a concurrent and prolonged loss of both AC and Vital DC power. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of Vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both AC and DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC	Alternating Current
AOP	Abnormal Operating Procedure
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
CNMT	Containment
DC	Direct Current
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPG	Emergency Procedure Guideline
FEMA	Federal Emergency Management Agency
GE	General Emergency
HCL	Heat Capacity Limit
HPCI	High Pressure Coolant Injection
IC	Initiating Condition
ID	Inside Diameter
ISFSI	Independent Spent Fuel Storage Installation
Keff	Effective Neutron Multiplication Factor
LCO	Limiting Condition of Operation
LOCA	Loss of Coolant Accident
mR, mRem, mrem, mREM	milli-Roentgen Equivalent Man
MW	Megawatt
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NORAD	North American Aerospace Defense Command
NOUE	Notification Of Unusual Event
NUMARC[1]	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
OCA	Owner Controlled Area
ODAM	Offsite Dose Assessment Manual
PA	Protected Area
PAG	Protective Action Guideline
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
PSIG	Pounds per Square Inch Gauge
R	Roentgen
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
Rem, rem, REM	Roentgen Equivalent Man

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[1] NUMARC was a predecessor organization of the Nuclear Energy Institute (NEI).

RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SCBA	Self-Contained Breathing Apparatus
SPDS	Safety Parameter Display System
TEDE	Total Effective Dose Equivalent
TAF	Top of Active Fuel
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report

APPENDIX B – DEFINITIONS

The following definitions are taken from Title 10, Code of Federal Regulations, and related regulatory guidance documents.

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

General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Notification of Unusual Event (NOUE): Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

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

The following are key terms necessary for overall understanding the DAEC emergency classification scheme.

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

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

Notification of Unusual Event (NOUE)
Alert
Site Area Emergency (SAE)
General Emergency (GE)

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

Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. This corresponds to the pressure boundary for the Dry Shielded Canister (DSC) shell (including the inner bottom cover plate) base metal and associated confinement boundary welds.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

DESIGN BASIS EARTHQUAKE (DBE): A DBE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

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

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

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

OPERATING BASIS EARTHQUAKE (OBE): An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.

OWNER CONTROLLED AREA: This term is typically taken to mean the site property owned by or otherwise under the control of the licensee.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY: Includes all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the Company. UFSAR Figure 1.2-1 identifies the DAEC SITE BOUNDARY.

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

ATTACHMENT 3

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO

LICENSE AMENDMENT REQUEST TSCR-166

UPDATED DEVIATIONS AND DIFFERENCES MATRIX

100 pages follow

UPDATED DAEC DEVIATIONS AND DIFFERENCES MATRIX

TABLE OF CONTENTS

GENERAL COMMENTS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
GLOBAL #1	References to NEI 99-01	Replaced with DAEC	Difference	Convert generic guidance to DAEC specific.	None
GLOBAL #2	Effective date	Replaced with **TBD, 2018**	Difference	Convert generic guidance to DAEC specific.	None
GLOBAL #3	Defined terms in Appendix B; Title Case	Defined terms in Appendix B; Upper Case	Difference	All defined terms in Appendix B used in the document are in upper case (CAPs) to indicate that the terms are defined.	None
GLOBAL #4	PWR specific references	PWR references removed	Difference	DAEC is a BWR	None
GLOBAL #5	Recognition Category A-Abnormal Radiation Levels/Radiological Effluent category and Emergency Action Levels; AU, AA, AS, and AG	Recognition Category R-Abnormal Radiation Levels/Radiological Effluent category and Emergency Action Levels; RU, RA, RS, and RG	Difference	DAEC implemented the optional designation of "R" for radiological related items to maintain continuity with previous practice at DAEC.	None
GLOBAL #6	Permanently Defueled Section	Deleted references to Permanently Defueled Station	Difference	Not Applicable to DAEC	None
GLOBAL #7	Acknowledgments, Notice and Executive Summary	Deleted	Difference	Not Applicable to DAEC	None
GLOBAL #8	Parameters or indications listed in EALs	Some parameters or indications listed in EALs were placed in tables or bulletized lists.	Difference	Tables or bullets were created to present DAEC-specific information in a manner familiar to and desired by scheme users.	None
GLOBAL #9	Site specific information or indication statements	"Site specific information or indications" were replaced with DAEC-specific information or indications where applicable.	Difference	Compliance with intent of the guidance.	None
GLOBAL #10	Operating Mode Applicability lists mode names (i.e., Power Operation, Startup)	Operating Mode Applicability lists mode numbers (i.e., 1, 2, etc.)	Difference	Mode numbers used for consistency with DAEC procedures and training.	None
GLOBAL #11	Developer's Notes	Developer's Notes deleted	Difference	Developer's notes are not reflected in the implementation of the EALs.	None
GLOBAL #12	Example EAL statement	"Example" deleted from statement	Difference	In adopting the EAL, the "example" status is no longer applicable.	None
GLOBAL #13	The following terms: "all, any, or, either" are sometimes capitalized and/or bolded in ICs and EALs	Consistently capitalized and bolded the following terms: "ALL, ANY, OR, EITHER" in ICs and EALs.	Difference	Capitalized and bolded conditional terms in ICs and EALs for consistency based on user feedback.	None
GLOBAL #14	Defined terms are only listed in APPENDIX B - DEFINITIONS	Defined terms are also listed as in separate section of each IC/EAL where the terms are used.	Difference	Aid to the user to present all needed information within the same section of the Basis document.	None
GLOBAL #15	Term "emergency buses"	Replaced with "essential buses"	Difference	Changed to reflect DAEC nomenclature	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
COVER PAGE	Development of Emergency Action Levels for Non-Passive Reactors	Duane Arnold Emergency Action Level Technical Bases Document	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
Introduction	Acknowledgments, Notice and Executive Summary	Deleted	Difference	Not Applicable to DAEC	None
TOC	1. Regulatory Background	1. Basis for Emergency Action Levels	Difference	Title change	None
TOC	1.1 Operating Reactors	1.1 Regulatory Background	Difference	Title change	None
TOC	1.2 Permanently Defueled Station	Deleted section	Difference	Not Applicable to DAEC	None
TOC	1.3 Independent Spent Fuel Storage Installation (ISFSI)	1.2 Independent Spent Fuel Storage Installation (ISFSI)	Difference	Re-numbered	None
TOC	1.4 NRC Order EA-12-051	1.3 NRC Order EA-12-051	Difference	Re-numbered	None
TOC	1.5 Applicability of Advance and Small Modular Reactor Designs	Deleted section	Difference	Not Applicable to DAEC	None
TOC	3.Design of the NEI 99-01 Emergency Classification Scheme	3. Design of the DAEC Emergency Classification Scheme	Difference	Title Change	None
TOC	3.3 NSSS Design Differences	Deleted section	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
TOC	3.4 Organization and Presentation of Generic Information	Changed to 3.3 DAEC 3.4 Organization and Presentation of Generic Information	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
TOC	4.0 Site-Specific Scheme Development	4.0 DAEC Scheme Development	Difference	Title change	None
TOC	4.4; 4.5; 4.6; 4.8	Deleted sections	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
TOC	4.7 Developer and User Feedback				None
TOC	Appendix C-Permanently Defueled Station ICs/EALs	Deleted section	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
1.1	Regulatory Background	Regulatory Background	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document and removed developer information	None
1.2	Permanently Defueled Station	Section deleted	Difference	Not Applicable to DAEC	None
1.3	1.3 Independent Spent Fuel Storage Installation (ISFSI)	1.2 Independent Spent Fuel Storage Installation (ISFSI)	Difference	Re-numbered section.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
1.4	1.4 NRC Order EA-12-051	1.3 NRC Order EA-12-051	Difference	Re-numbered and removed wording to add these readings (DAEC installation completed).	None
1.5	Applicability to Advanced and Small Modular Reactor Designs	Section deleted	Difference	Not Applicable to DAEC	None
2	KEY TERMINOLOGY USED IN NEI 99-01	KEY TERMINOLOGY USED IN DAEC EAL SCHEME	Difference	Minor changes to reflect DAEC-specific implementation.	None
3	DESIGN OF THE NEI 99-01 EMERGENCY CLASSIFICATION SCHEME	DESIGN OF THE DAEC EMERGENCY CLASSIFICATION SCHEME	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
3.1	Assignment of Emergency Classification Levels (ECLs)	Assignment of Emergency Classification Levels (ECLs)	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document, removed references to PWRs, and removed developer information.	None
3.2	Types of Initiating Conditions and Emergency Action Levels	Types of Initiating Conditions and Emergency Action Levels	Verbatim		None
3.3	Text referring to NSSS design differences for various types or plants; Developer guidance	Deleted	Difference	Guidance is now DAEC specific	None
3.4	Organization and Presentation of Generic Information	DAEC-Specific Organization and Presentation of Generic Information	Difference	Renumbered to 3.3, made DAEC-specific, and deleted developer information	None
3.5	Mode of Applicability Matrix; Typical BWR Operating Modes	Deleted "Permanently Defueled" section of matrix; replaced Typical BWR Operating Modes with DAEC-specific Operating Modes	Difference	Renumbered to 3.4, removed PWR information, removed permanently defueled, and inserted DAEC Operating Modes to comply with the document intent.	V1
4	Site Specific Scheme Development Guidance	Development of the DAEC Emergency Classification Scheme	Difference	Updated to reflect DAEC specific scheme development process.	None
5	GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS	GUIDANCE ON USING THE DAEC EALS	Difference	Guidance is now DAEC specific	None
6 - 11	Recognition Category IC/EAL Matrixes	removed	Difference	Matrixes were intended for use by EAL developers. Inclusion in licensee scheme is not desired.	None

ABNORMAL RAD LEVELS / RADIOACTIVE EFFLUENT ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AU1	Recognition Category: AU1	RU1	Difference	Global Comment #5	None
	Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.	Release of gaseous or liquid radioactivity greater than 2 times the ODAM limits for 60 minutes or longer.	Difference	Global Comment #9	None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AU1 (cont.)	(1) Reading on ANY effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer: (site-specific monitor list and threshold values corresponding to 2 times the controlling document limits)	(1) Reading on **ANY** Table R-1 effluent radiation monitor greater than column "NOUE" for 60 minutes or longer: [inserted Table R-1 of DAEC-specific radiation monitors and threshold values]	Difference	See Global Comments #8, 9, 12, & 13. Reworded EAL statement to remove operator confusion as to whether they needed to multiply the values of the following table by 2 or if the value provided already was 2X. Wording now matches wording of RS1 and RG1 allowing for easier operator progression through the EALs.	V2
	(2) Reading on ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	(2) Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	Difference	Global Comment #13	None
	(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.	(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODAM limits for 60 minutes or longer.	Difference	Global Comment #9	None
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AU2	Recognition Category: AU2	RU2	Difference	Global Comment #5 & 14	None
	Initiating Condition: UNPLANNED loss of water level above irradiated fuel.	UNPLANNED loss of water level above irradiated fuel.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None
	(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following: (site-specific level indications). **AND**	(1) a UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: • Report to control room (visual observation) • Fuel pool level indication (LI-3413) less than 36 feet and lowering • WR GEMAC Floodup indication (LI-4541) coming on scale **AND**	Difference	Global Comment #9, 12 & 13	V3

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AU2 (cont.)	b. UNPLANNED increase in area radiation levels as indicated by ANY of the following radiation monitors. (site-specific list of area radiation monitors)	b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors. • Spent Fuel Pool Area, RI-9178 • North Refuel Floor, RI-9163 • New Fuel Vault Area, RI-9153 • South Refuel Floor, RI-9164 • NW Drywell Area Hi Range Rad Monitor, RIM-9184A • South Drywell Area Hi Range Rad Monitor, RIM-9184B	Difference	Global Comments #9 & 13 Intent and meaning of the EALs are not altered.	V4

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AA1	Recognition Category: AA1	RA1	Difference	Global Comment #5 & 14	None
	Initiating condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.	Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None
	(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	(1) Reading on **ANY** Table R-1 radiation monitor greater than column "Alert" for 15 minutes or longer: [inserted Table R-1 of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V5
	(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]	Difference	Global Comment #9 Added bracketed 'Preferred' to reinforce the 4th Note of the IC	None
	(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.	(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.	Difference	Global Comment #9	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AA1 (cont.)	(4) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	(4) Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AA2	Recognition Category: AA2	RA2	Difference	Global Comment #5 & 14	None
	Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.	Significant lowering of water level above, or damage to, irradiated fuel.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None
	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.	Verbatim		None
	(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following radiation monitors: (site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)	(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by Hi Rad alarm for **ANY** of the following ARMs: • Spent Fuel Pool Area, RI-9178 • North Refuel Floor, RI-9163 • New Fuel Vault Area, RI-9153 • South Refuel Floor, RI-9164 **OR** Reading greater than 5 R/hr on ANY of the following radiation monitors (in Mode 5 only): • NW Drywell Area Hi Range Rad Monitor, RIM-9184A • South Drywell Area Hi Range Rad Monitor, RIM-9184B	Difference	Global Comment #8, 9, 12 & 13 Threshold values for the Drywell monitors are only applicable in Mode 5 since the calculated radiation levels from damage to irradiated fuel would be masked by the typical background levels on these monitors during plant operation, and mechanical damage to a fuel assembly in the vessel can only happen with the reactor head removed (Mode 5).	V6
	(3) Lowering of spent fuel pool level to (site-specific Level 2 value). [See Developer Notes	(3) Lowering of spent fuel pool level to 25.17 feet	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V7

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AA3	Recognition Category: AA3	RA3	Difference	Global Comment #5 & 14	None
	Initiating Condition: Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.	Radiation levels that impede access to areas necessary for normal plant operation.	Difference	Reworded IC to reflect non-applicability of EAL #2.	None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Dose rate greater than 15 mR/hr in ANY of the following areas: • Control Room • Central Alarm Station • (other site-specific areas/rooms)	(1) Dose rate greater than 15 mR/hr in **ANY** of the following areas: • Control Room ARM (RM-9162) • Central Alarm Station (by survey)	Difference	Global Comment #9, 12 & 13	None
	(2) An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified)	Not used at DAEC	Difference	EALs RA3 and HA5 are not applicable to DAEC because an evaluation has shown that there are no rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. All areas outside the Control Room that contain equipment necessary for normal plant operation, cooldown and shutdown do not require physical access to operate. Intent and meaning of the EALs are not altered.	V8

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AS1	Recognition Category: AS1	RS1	Difference	Global Comment #5 & 14	None
	Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	(1) Reading on **ANY** Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer. [inserted Table R-1 of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V9
	(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]	Difference	Global Comment #3 & 9 Added bracketed 'Preferred' to reinforce the 4th Note of the IC	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AS1 (cont.)	(3) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.	(3) Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.	Difference	Global Comment #3, 9, & 13 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AS2	Recognition Category: AS2	RS2	Difference	Global Comment #5	None
	Initiating Condition: Spent fuel pool level at (site-specific Level 3 description).	Spent fuel pool level at 16.36 feet	Difference	Global Comment #9	V10
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Lowering of spent fuel pool level to (site-specific Level 3 value).	(1) Lowering of spent fuel pool level to 16.36 feet	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V10

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AG1	Recognition Category: AG1	RG1	Difference	Global Comment #5 & 14	None
	Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.	Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	(1) Reading on ANY Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer. [inserted Table R-1 of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V9
	(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]	Difference	Global Comment #3 & 9 Added bracketed 'Preferred' to reinforce the 4th Note of the IC	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AG1 (cont.)	(3) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.	(3) Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.	Difference	Global Comment #3 & 9	None
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AG2	Recognition Category: AG2	RG2	Difference	Global Comment #5	None
	Initiating Condition: Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer.	Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.	Difference	Global Comment #9	V10
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer.	(1) Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V10

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU1	Recognition Category: CU1	CU1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: UNPLANNED loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory for 15 minutes or longer.	UNPLANNED loss of RPV inventory for 15 minutes or longer	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [PWR] or RPV [BWR]) level less than a required lower limit for 15 minutes or longer.	(1) UNPLANNED loss of reactor coolant results in RPV level less than a required lower limit for 15 minutes or longer.	Difference	Global Comment #4 & 12	None
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored. AND	(2) a. RPV level cannot be monitored. **AND**	Difference	Global Comment #4	None
	b. UNPLANNED increase in (site-specific sump and/or tank) levels.	b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None
CU2	Recognition Category: CU2	CU2	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Loss of all but one AC power source to essential buses for 15 minutes or longer.	Difference	Global comment #15	None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 4, 5, Defueled	Difference	Global Comment #10	None
	(1) a. AC power capability to (site-specific emergency buses) is reduced to a	(1) a. AC power capability to 1A3 and 1A4 buses is reduced to a single power	Difference	Global Comment #9, 12, & 13	V11

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
	single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.	source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.		Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU3	Recognition Category: CU3	CU3	Verbatim	Global Comment #11, 14	None
	Initiating Condition: UNPLANNED increase in RCS temperature.	UNPLANNED increase in RCS temperature.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit).	(1) UNPLANNED increase in RCS temperature to greater than 212°F	Difference	Global Comment #9 & 12	V1
	(2) Loss of ALL RCS temperature and (reactor vessel/RCS [PWR] or RPV [BWR]) level indication for 15 minutes or longer.	(2) Loss of **ALL** RCS temperature and RPV level indication for 15 minutes or longer	Difference	Global Comment #4 & 13 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU4	Recognition Category: CU4	CU4	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of Vital DC power for 15 minutes or longer.	Loss of Vital DC power for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer.	(1) Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer	Difference	Global Comment #9, 12, 13 Intent and meaning of the EALs are not altered.	V12

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU5	Recognition Category: CU5	CU5	Verbatim		None
	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Loss of all onsite or offsite communications capabilities.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 4, 5, Defueled	Difference	Global Comment #10	None
	(1) Loss of ALL of the following onsite communication methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following onsite communication methods: • Plant Operations Radio System • In-Plant Phone System • Plant Paging System (Gaitronics)	Difference	Global Comment #9, 12 & 13	V13
	(2) Loss of ALL of the following ORO communications methods: (site-specific list of communications methods)	(2) Loss of **ALL** of the following offsite response organization communications methods: • DAEC All-Call phone • All telephone lines (PBX and commercial) • Cell Phones (including fixed cell phone system) • Control Room fixed satellite phone system • FTS Phone system	Difference	Global Comment #9 & 13	V13 V14

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU5 (cont.)	(3) Loss of ALL of the following NRC communications methods: (site-specific list of communications methods)	(3) Loss of **ALL** of the following NRC communications methods: • FTS Phone system • All telephone lines (PBX and commercial) • Cell Phones (including fixed cell phone system) • Control Room fixed satellite phone system	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	V13

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CA1	Recognition Category: CA1	CA1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory.	Loss of RPV inventory.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory as indicated by level less than (site-specific level).	(1) Loss of RPV inventory as indicated by level less than 119.5 inches	Difference	Global Comment #4, 9 & 12	V15
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 15 minutes or longer	(2) a. RPV level cannot be monitored for 15 minutes or longer	Difference	Global Comment #4	None
	AND b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory.	**AND** b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool due to a loss of RPV inventory.	Difference	Global Comment #4, 9 & 13 Intent and meaning of the EALs are not altered.	None

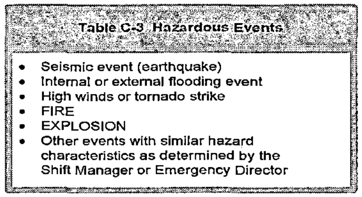
Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CA2	Recognition Category: CA2	CA2	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Loss of all offsite and all onsite AC power to essential buses for 15 minutes or longer.	Difference	Global Comment #15	None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 4, 5, Defueled	Difference	Global Comment #10	None
	(1) Loss of ALL offsite and ALL onsite AC Power to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite and **ALL** onsite AC Power to 1A3 and 1A4 for 15 minutes or longer.	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	V12

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="6">CA3</td><td>Recognition Category: CA3</td><td>CA3</td><td>Verbatim</td><td>Global Comment #11, 14</td><td>None</td></tr>
<tr><td>Initiating Condition: Inability to maintain the plant in cold shutdown.</td><td>Inability to maintain the plant in cold shutdown.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: Cold Shutdown, Refueling</td><td>Operating Mode Applicability: 4, 5</td><td>Difference</td><td>Global Comment #10</td><td>None</td></tr>
<tr><td>(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in the following table.</td><td>(1) UNPLANNED increase in RCS temperature to greater than 212°F for greater than the duration specified in Table C-2.</td><td>Difference</td><td>Global Comment #9 & 12</td><td>V1</td></tr>
<tr><td>
<table>
<tr><th colspan="3">Table: RCS Heat-up Duration Thresholds</th></tr>
<tr><th>RCS Status</th><th>Containment Closure Status</th><th>Heat-up Duration</th></tr>
<tr><td>Intact (but not at reduced inventory [PWR])</td><td>Not applicable</td><td>60 minutes*</td></tr>
<tr><td rowspan="2">Not intact (or at reduced inventory [PWR])</td><td>Established</td><td>20 minutes*</td></tr>
<tr><td>Not Established</td><td>0 minutes</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.</td></tr>
</table>
</td><td>
<table>
<tr><th colspan="3">Table C-2 RCS Heat-up Duration Thr</th></tr>
<tr><th>RCS Integrity</th><th>Containment Closure Status</th><th>Heat-up Duration</th></tr>
<tr><td>Intact</td><td>Not Applicable</td><td>60 minutes*</td></tr>
<tr><td rowspan="2">Not intact</td><td>Established</td><td>20 minutes*</td></tr>
<tr><td>Not Established</td><td>0 minutes</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operatior
frame and RCS temperature is being reduced,
applicable.</td></tr>
</table>
</td><td>Difference</td><td>Global Comment #4
Changed “RCS Status” to “RCS Integrity” to match current site nomenclature</td><td>None</td></tr>
<tr><td>(2) UNPLANNED RCS pressure increase greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [PWR])</td><td>(2) UNPLANNED RCS pressure increase greater than 10 psig due to a loss of RCS cooling.</td><td>Difference</td><td>Global Comment #4 & 9
Added “due to a loss of RCS cooling” to clarify the intent of the EAL

Intent and meaning of the EALs are not altered.</td><td>V16</td></tr>
</table>

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CA6	Recognition Category: CA6	CA6	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager	(1) a. The occurrence of ANY of the Table C-3 hazardous events:  **Table C-3 Hazardous Events** • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director	Difference	Global Comment #9, 12 & 13	None

CA6 (cont.)	**AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.	**AND** b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode. **AND**	Deviation	Adopted the revised EAL wording provided in approved EAL FAQ 2016-02.	V17
	OR 1. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	2. **EITHER** of the following: • Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or • The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.	Deviation	Adopted the revised EAL wording provided in approved EAL FAQ 2016-02.	V17
			Difference	Added the following clarification to the Basis from EALFAQ 2018-04: An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under SA8 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement. Intent and meaning of the EALs are not altered.	V18

DAEC DEVIATIONS AND DIFFERENCES MATRIX

CS1	Recognition Category: CS1	CS1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory affecting core decay heat removal capability.	Loss of reactor vessel/RCS inventory affecting core decay heat removal capability.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level).	(1) a. CONTAINMENT CLOSURE not established. **AND** b. RPV level less than +64 inches	Difference	Global Comment #9 & 12	V19
	(2) a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level).	(2) a. CONTAINMENT CLOSURE established. **AND** b. RPV level less than +15 inches	Difference	Global Comment #4 & 9	V19

CS1 (cont.)	(3) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer. **AND**	(3) a. RPV level cannot be monitored for 30 minutes or longer. **AND**	Difference	Global Comment #4	None
	b. Core uncovery is indicated by **ANY** of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [PWR] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications)	b. Core uncovery is indicated by **ANY** of the following: • Drywell Monitor (9184A/B) reading greater than 5.0 R/hr • UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery	Difference	Global Comment #9 &13	V6
				Intent and meaning of the EALs are not altered.	

CG1	Recognition Category: CG1	CG1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory affecting fuel clad integrity with containment challenged.	Loss of reactor vessel/RCS inventory affecting fuel clad integrity with containment challenged.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level) for 30 minutes or longer. **AND** b. ANY indication from the Containment Challenge Table (see below).	(1) a. RPV level less than +15 inches for 30 minutes or longer. **AND** b. ANY indication from the Containment Challenge Table (see below).	Difference	Global Comment #4, 9, 12 & 13	V19
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer.	(2) a. RPV level cannot be monitored for 30 minutes or longer.	Difference	Global Comment #4	None

	AND b. Core uncovery is indicated by ANY of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [PWR] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery **AND**	**AND** b. Core uncovery is indicated by **ANY** of the following: • Drywell Monitor (9184A/B) reading greater than 5.0 R/hr • Erratic source range monitor indication • UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery	Difference	Global Comment #8, 9 & 13	V6
	c. **ANY** indication from the Containment Challenge Table (see below).	**AND** c. **ANY** indication from the Secondary Containment Challenge Table C-1.	Difference	Global Comment #9	None
	Containment Challenge Table ONTAINMENT CLOSURE not established* Explosive mixture) exists inside containment NPLANNED increase in containment pressure econdary containment radiation monitor reading site specific value) [BWR]	**Table C-1 Containment Challenge Ta** • CONTAINMENT CLOSURE not established • Drywell Hydrogen or Torus Hydrogen grea **AND** Drywell Oxygen or Torus Oxygen gre • UNPLANNED increase in containment pre • Secondary containment radiation monito safe operating limits (MSOL) of EOP 3, Ta	Difference	Global Comment #9	V20 V21
	* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.	*If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.	Verbatim	Intent and meaning of the EALs are not altered.	

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="4">E-HU1</td><td>Recognition Category: E-HU1</td><td>E-HU1</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.</td><td>Damage to a loaded cask CONFINEMENT BOUNDARY.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: All</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.</td><td>(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by a radiation reading greater than the values shown on Table E-1 on the spent fuel cask.
<table>
<tr><th colspan="2">Table E-1 Cask Dose Rates</th></tr>
<tr><td>61BT DSC</td><td>800 mrem/hr</td></tr>
<tr><td>3 feet from HSM Surface</td><td>200 mrem/hr</td></tr>
<tr><td>Outside HSM Door – Centerline of DSC</td><td>40 mrem/hr</td></tr>
</table></td><td>Difference</td><td>Global Comment #8, 9, 12 & 14

Intent and meaning of the EALs are not altered.</td><td>V22</td></tr>
</table>

FISSION PRODUCT BARRIER ICS/EALS

The following section is configured in a manner that is different from the Fission Product Barrier Tables in the DAEC EAL Technical Bases Document. Where the Technical Bases Document evaluates all three fission product barriers simultaneously for a specific sub-category, this matrix presents each fission product barrier individually for all sub-categories. The significance of this presentation is that where the fission product barrier table in the Technical Bases Document moves vertically through the sub-categories, this matrix moves horizontally.

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th colspan="6">Fission Product Barrier Emergency Classifications</th><th></th></tr>
<tr><th colspan="3">NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td colspan="3">Table 9-F-1: Recognition Category "F" Initiating Condition Matrix</td><td rowspan="4">Deleted</td><td rowspan="4">Difference</td><td rowspan="4">Deleted per developer note. Mode applicability carried over onto Table 9-F EAL listing.
Global Comment #11</td><td rowspan="4">None</td></tr>
<tr><td>Alert</td><td>Site Area Emergency</td><td>General Emergency</td></tr>
<tr><td>Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.</td><td>Loss or Potential Loss of any two barriers.</td><td>Loss of any two barriers and Loss or Potential Loss of the third barrier.</td></tr>
<tr><td><i>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</i></td><td><i>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</i></td><td><i>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</i></td></tr>
<tr><td colspan="3">Table 9-F-2: BWR EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Table F-1: DAEC EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Difference</td><td>Renumbered and re-labeled due to deletion of Tables 9-F-1 & 3.
Added Global Comment #9</td><td>None</td></tr>
<tr><td colspan="3">Table 9-F-3: PWR EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Deleted</td><td>Difference</td><td>Global Comment #4</td><td>None</td></tr>
<tr><td colspan="3">Basis Information For BWR EAL Fission Product Barrier Table 9-F Developer Notes.</td><td>Deleted Developer Notes</td><td>Difference</td><td>Transform generic NEI 99-01 guidance into DAEC-specific application.</td><td>None</td></tr>
<tr><td colspan="3">Figure 9-F-4: PWR Containment Integrity or Bypass Example</td><td>Deleted</td><td>Difference</td><td>Global Comment #4</td><td>None</td></tr>
</table>

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Fuel Clad Barrier						
Table 9-F	NEI 99-01 Rev. 6		DAEC		Change	Justification
Sub-Category	Loss	Potential Loss	Loss	Potential Loss		
1. RCS Activity	A. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).	Not Applicable	A. Coolant activity greater than 300 µCi/gm dose equivalent I-131	Not Applicable	Difference	General Comment #9
2. RPV Water Level	A. Primary containment flooding required.	A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) or cannot be determined.	A. SAG entry is required.	A. RPV water level cannot be restored and maintained above +15 inches **OR** cannot be determined.	Difference	EPFAQ 2015-004 V15 General Comment #9, 13
3. Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Verbatim	None
4. Primary Containment Radiation	A. Primary containment radiation monitor reading greater than (site-specific value).	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 2000 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 200 R/hr	Not Applicable	Difference	V23 Global Comment #9
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	A. Fuel damage assessment indicates at least 5% fuel clad damage.	Not Applicable	Difference	Global Comment #9 Core damage assessment procedure.

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Fuel Clad Barrier						
Table 9-F	**NEI 99-01 Rev. 6**		**DAEC**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Verbatim	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of RCS Barrier						
Table 9-F	**NEI 99-01 Rev. 6**		**DAEC**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
1. Primary Containment Pressure Renamed to **1. Primary Containment Conditions**	A. Primary containment pressure greater than (site-specific value) due to RCS leakage.	Not Applicable	A. Primary containment pressure greater than 2 psig due to RCS leakage.	Not Applicable	Difference	V24 Global Comment #9
2. RPV Water Level	A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) or cannot be determined.	Not Applicable	A. RPV water level cannot be restored and maintained above +15 inches **OR** cannot be determined.	Not Applicable	Difference	V19 Global Comment #9, 13
3. RCS Leak Rate	A. UNISOLABLE break in **ANY** of the following: (site-specific systems with potential for high-energy line breaks) **OR** B. Emergency RPV Depressurization.	A. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following: 1. Max Normal Operating Temperature **OR** 2. Max Normal Operating Area Radiation Level.	A. UNISOLABLE break in Main Steam, HPCI, Feedwater, RWCU, or RCIC as indicated by the failure of both isolation valves in **ANY** one line to close **AND EITHER:** • High MSL flow or steam tunnel temperature annunciators **OR** • Direct report of steam release **OR** B. Emergency RPV Depressurization required.	A. UNISOLABLE primary system leakage that results in exceeding the Max Normal Operating Limit (MNOL) of EOP 3, Table 6 for **EITHER** of the following: • Temperature **OR** • Radiation Level	Difference	V25 Global Comment #9 Added site-specific indication of an unisolable steam line break which includes failure of both isolation valves to LOSS 3.A.

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of RCS Barrier						
4. Primary Containment Radiation	A. Primary containment radiation monitor reading greater than (site-specific value).	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5 R/hr after reactor shutdown	Not Applicable	Difference	Global Comment #9 V23
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Not Applicable	Not Applicable	Difference	Global Comment #9
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Verbatim	None

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**DAEC**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
1. Primary Containment Conditions	A. UNPLANNED rapid drop in primary containment pressure following primary containment pressure rise **OR** B. Primary containment pressure response not consistent with LOCA conditions.	A. Primary containment pressure greater than (site-specific value) **OR** B. (site-specific explosive mixture) exists inside primary containment **OR** C. HCTL exceeded.	A. UNPLANNED rapid drop in Drywell pressure following Drywell pressure rise **OR** B. Drywell pressure response not consistent with LOCA conditions. **OR** C. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal **OR** D. Intentional primary containment venting per EOPs	A. Torus pressure greater than 53 psig **OR** B. Drywell or Torus H2 cannot be determined to be less than 6% and Drywell **OR** Torus O2 cannot be determined to be less than 5% **OR** C. HCL (Graph 4 of EOP 2) exceeded.	Difference	Global Comment #9 V20 V26 V27 Primary Containment Isolation Failure Loss 3.A and 3.B moved to sub-category 1 "Primary Containment Conditions" as Losses 1.C and 1.D to consolidate concepts into single sub-category
2. RPV Water Level	Not Applicable	A. Primary containment flooding required.	Not Applicable	A. SAG entry is required.	Difference	EPFAQ 2015-004

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	NEI 99-01 Rev. 6		DAEC		Change	Justification
Sub-Category	Loss	Potential Loss	Loss	Potential Loss		
3. Primary Containment Isolation Failure	A. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal **OR** B. Intentional primary containment venting per EOPs **OR** C. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: 1. Max Safe Operating Temperature. **OR** 2. Max Safe Operating Area Radiation Level.	Not Applicable	A. UNISOLABLE primary system leakage that results in exceeding the Max Safe Operating Limit (MSOL) of EOP 3, Table 6 for **EITHER** of the following: • Temperature **OR** • Radiation Level	Not Applicable	Difference	Global Comment #9 V28 Primary Containment Isolation Failure Loss 3.A and 3.B moved to sub-category 1 "Primary Containment Conditions" as Losses 1.C and 1.D to consolidate concepts into single sub-category
4. Primary Containment Radiation	Not Applicable	A. Primary containment radiation monitor reading greater than (site-specific value).	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5000 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 500 R/hr	Difference	Global Comment #9 V23

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**DAEC**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Not Applicable	A. Fuel damage assessment indicates at least 20% fuel clad damage.	Difference	Global Comment #9 Core damage assessment procedure.
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	B. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	**C.** **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	D. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	Verbatim	None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU1	Recognition Category: HU1	HU1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Confirmed SECURITY CONDITION or threat.	Confirmed SECURITY CONDITION or threat.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).	(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by DAEC Security Shift Supervision.	Difference	Global Comment #9 & 12	None
	(2) Notification of a credible security threat directed at the site.	(2) Notification of a credible security threat directed at DAEC.	Difference	Global Comment #9	None
	(3) A validated notification from the NRC providing information of an aircraft threat.	(3) A validated notification from the NRC providing information of an aircraft threat.	Verbatim	None Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU2	Recognition Category: HU2	HU2	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Seismic event greater than OBE levels.	Seismic event greater than OBE levels.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits)	(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by receipt of the Amber Operating Basis Earthquake Light and the wailing seismic alarm on 1C35.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V29

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU3	Recognition Category: HU3	HU3	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Hazardous event.	Hazardous event.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A tornado strike within the PROTECTED AREA.	(1) A tornado strike within the PROTECTED AREA.	Verbatim	Global Comment #12	None
	(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	Verbatim		None
	(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	Verbatim		None
	(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	Verbatim		None
	(5) (Site-specific list of natural or technological hazard events)		Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU4	Recognition Category: HU4	HU4	Verbatim	Global Comment #11, 14	None
	Initiating Condition: FIRE potentially degrading the level of safety of the plant.	FIRE potentially degrading the level of safety of the plant.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) a. A FIRE is NOT extinguished within 15-minutes of ANY of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND**	(1) a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND**	Difference	Global Comment #12 & 13	None
	b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas)	b. The FIRE is located within **ANY** Table H-1 plant rooms or areas. **Table H-1 Fire Areas** • 1G31 DG and Day Tank Rooms • 1G21 DG and Day Tank Rooms • Battery Rooms • Essential Switchgear Rooms • Cable Spreading Room • Torus Room • Intake Structure • Pumphouse • Drywell • Torus • NE, NW, SE Corner Rooms • HPCI Room • RCIC Room • RHR Valve Room • North CRD Area • South CRD Area • CSTs • Control Building • Remote Shutdown Panel 1C388 Area • Panel 1C55/56 Area • SBGT Room	Difference	Global Comment #8, 9, & 13	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU4 (cont.)	(2) a. Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas) **AND**	(2) a. Receipt of a single fire alarm with no other indications of a FIRE. **AND** b. The FIRE is located within **ANY** Table H-1 plant rooms or areas. **AND**	Difference	Global Comment #8, 9 & 13	None
	c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.	c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.	Verbatim	N/A	None
	(3) A FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.	(3) A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60 minutes of the initial report, alarm or indication.	Difference	Global Comment #9	None
	(4) A FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	(4) A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	Difference	Global Comment #9	None
				Basis revised to include NFPA-805 in the discussion of Appendix R basis for the EAL thresholds. Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU7	Recognition Category: HU7	HU7	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO) UE.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE.	Difference	NOUE versus (NO)UE, DAEC uses the full NOUE term	None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	Verbatim	Global Comment #3, 12, 14	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HA1	Recognition Category: HA1	HA1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision). (2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the DAEC Security Shift Supervision. (2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	Difference Verbatim	Global Comment #9, 12, 14 Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HA5	Recognition Category: HA5	Not used at DAEC	Difference	EALs RA3 and HA5 are not applicable to DAEC because an evaluation has shown that there are no rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. All areas outside the Control Room that contain equipment necessary for normal plant operation, cooldown and shutdown do not require physical access to operate.	V8
	Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.	Not used at DAEC	Difference		None
	Operating Mode Applicability: All	Not used at DAEC	Difference		None
	(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified) **AND** b. Entry into the room or area is prohibited or impeded.	Not used at DAEC	Difference		None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HA6	Recognition Category: HA6	HA5	Difference	Renumbered to align with other similar ICs	None
	Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.	Control Room evacuation resulting in transfer of plant control to alternate locations.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	(1) An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V30

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HA7	Recognition Category: HA7	HA6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HS1	Recognition Category: HS1	HS1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.	HOSTILE ACTION within the PROTECTED AREA.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HS6	Recognition Category: HS6	HS5	Difference	Renumbered to align with other similar ICs	None
	Initiating Condition: Inability to control a key safety function from outside the Control Room.	Inability to control a key safety function from outside the Control Room.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that (site specific number of) minutes has been exceeded, or will likely be exceeded.	**Note:** The Emergency Director should declare the Site Area Emergency promptly upon determining that 20 minutes has been exceeded, or will likely be exceeded.		Global Comment #9	V30
	(1) a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	(1) a. An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).	Difference	Global Comment #9, 12	None
	AND b. Control of **ANY** of the following key safety functions is not reestablished within (site-specific number of minutes). • Reactivity control • Core cooling [PWR] / RPV water level [BWR] • RCS heat removal	**AND** b. Control of **ANY** of the following key safety functions is not reestablished within 20 minutes. • Reactivity control • RPV water level • RCS heat removal	Difference	Global Comment #4, 9 Intent and meaning of the EALs are not altered.	V30

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HS7	Recognition Category: HS7	Recognition Category: HS6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: ALL	Verbatim		None
	(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.	(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.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HG1	Recognition Category: HG1	HG1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility.	HOSTILE ACTION resulting in loss of physical control of the facility.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.	Difference	Global Comment #9, 12	None
	AND b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity control • Core cooling [PWR] / RPV water level [BWR] • RCS heat removal	**AND** b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity control • RPV water level • RCS heat removal	Difference	Global Comment #4, 9	None
	OR 2. Damage to spent fuel has occurred or is IMMINENT.	**OR** 2. Damage to spent fuel has occurred or is IMMINENT.	Verbatim		None
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HG7	Recognition Category: HG7	HG6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

SYSTEM MALFUNCTION ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU1	Recognition Category: SU1	SU1	Verbatim		None
	Initiating Condition: Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.	Loss of ALL offsite AC power capability to essential buses for 15 minutes or longer.	Difference	Global Comment #15	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) Loss of **ALL** offsite AC power capability to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite AC power capability to 1A3 AND 1A4 buses for 15 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="5">SU2</td><td>Recognition Category: SU2</td><td>SU3</td><td>Difference</td><td>Global Comment #11, 14
Renumbered to align with other similar ICs</td><td>None</td></tr>
<tr><td>Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.</td><td>UNPLANNED loss of Control Room indications for 15 minutes or longer.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown</td><td>Operating Mode Applicability: 1, 2, 3</td><td>Difference</td><td>Global Comment #10</td><td>None</td></tr>
<tr><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.</td><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the Table S-1 parameters from within the Control Room for 15 minutes or longer.</td><td>Difference</td><td>Global Comment #12</td><td>None</td></tr>
<tr><td><table>
<tr><th>[BWR parameter list]</th><th>[PWR parameter list]</th></tr>
<tr><td>Reactor Power</td><td>Reactor Power</td></tr>
<tr><td>RPV Water Level</td><td>RCS Level</td></tr>
<tr><td>RPV Pressure</td><td>RCS Pressure</td></tr>
<tr><td>Primary Containment Pressure</td><td>In-Core/Core Exit Temperature</td></tr>
<tr><td>Suppression Pool Level</td><td>Levels in at least (site-specific number) two steam generators</td></tr>
<tr><td>Suppression Pool Temperature</td><td>Steam Generator Auxiliary or Emergency Feed Water Flow</td></tr>
</table></td><td>Table S-1 Safety System Parameters
• Reactor Power
• RPV Water Level
• RPV Pressure
• Primary Containment Pressure
• Suppression Pool Level
• Suppression Pool Temperature</td><td>Difference</td><td>Global Comment #4, 9
Intent and meaning of the EALs are not altered.</td><td>None</td></tr>
</table>

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU3	Recognition Category: SU3	SU4	Verbatim	Global Comment #11, 14R Renumbered IC to align with other similar ICs	None
	Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.	Reactor coolant activity greater than Technical Specification allowable limits.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) (Site-specific radiation monitor) reading greater than (site-specific value).	(1) Pretreatment Offgas System (RM-4104) Hi-Hi Radiation Alarm	Difference	Global Comment #9 & 12	None
	(2) Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	(2) Sample analysis indicates that reactor coolant specific activity is greater than 2.0 µCi/gm dose equivalent I-131 for 12 hours or longer.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V31

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU4	Recognition Category: SU4	SU5	Verbatim	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: RCS leakage for 15 minutes or longer.	RCS leakage for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.	(1) RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.	Difference	Global Comment #9 & 12	V32
	(2) RCS identified leakage greater than (site-specific value) for 15 minutes or longer.	(2) RCS identified leakage greater than 25 gpm for 15 minutes or longer.	Difference	Global Comment #9	V32
	(3) Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.	(3) Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.	Verbatim		None
				Intent and meaning of the EALs are not altered.	

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU5	Recognition Category: SU5	SU6	Verbatim	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor.	Automatic or manual scram fails to shutdown the reactor.	Difference	Global Comment #4	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	Global Comment #10 DAEC can be up to 12% power in STARTUP Mode, so Mode 2 applicability added	V33
	(1) a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic scram did not shutdown the reactor.	Difference	Global Comment #4 & 12	None
	AND b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	**AND** b. **ANY** of the following manual actions taken at 1C05 are successful in lowering reactor power below 5% power • Manual Scram Pushbuttons • Mode Switch to Shutdown • Alternate Rod Insertion (ARI)	Difference	Global Comment #9	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU5 (cont.)	(2) a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor.	(2) a. A manual scram did not shutdown the reactor.	Difference	Global Comment #4	None
	AND b. EITHER of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	**AND** b. 1. **EITHER** of the following subsequent manual actions taken at 1C05 <u>are successful</u> in lowering reactor power below 5% power • Manual Scram Pushbuttons • Mode Switch to Shutdown • Alternate Rod Insertion (ARI)	Difference	Global Comment #9	None
	OR 2. A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.	**OR** 2. A subsequent automatic scram is successful in shutting down the reactor.	Difference	Global Comment #4 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU6	Recognition Category: SU6	SU7	Verbatim	Global Comment #14 Renumbered to align with other similar ICs	None
	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Loss of **ALL** onsite or offsite communications capabilities.	Difference	Global Comment #13	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) Loss of ALL of the following Onsite communication methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following Onsite communication methods: • Plant Operations Radio System • In-Plant Phone System • Plant Paging System (Gaitronics)	Difference	Global Comment #9, 12 & 13	V16
	(2) Loss of ALL of the following ORO communications methods: (site-specific list of communications methods)	(2) Loss of **ALL** of the following offsite response organization communications methods: • DAEC All-Call phone • All telephone lines (PBX and commercial) • Cell Phones (including fixed cell phone system) • Control Room fixed satellite phone system • FTS Phone system	Difference	Global Comment #9 & 13	V13, V14

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU6 (cont.)	(3) Loss of ALL of the following NRC communications methods: (site-specific list of communications methods)	(4) Loss of **ALL** of the following NRC communications methods: • FTS Phone system • All telephone lines (PBX and commercial) • Cell Phones (including fixed cell phone system) • Control Room fixed satellite phone system	Difference	Global Comment #9 & 13 Intent and meaning of the EALs are not altered.	V13, V14

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU7	Recognition Category: SU7 Initiating Condition: Failure to isolate containment or loss of containment pressure control. [PWR] Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown (1) a. Failure of containment to isolate when required by an actuation signal. **AND** b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal. (1) a. Containment pressure greater than (site-specific pressure). **AND** b. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.	Not Applicable	Difference	Global Comment #4 This IC and EALs are only applicable to PWR plants.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA1	Recognition Category: SA1	SA1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer.	Difference	Global Comment #15	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer.	(1) a. AC power capability to 1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.	Difference	Global Comment #9, 12	None
	AND b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.	**AND** a. **ANY** additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.	Difference	Global Comment #13 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<thead>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
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<tr><td rowspan="5">SA2</td><td>Recognition Category: SA2</td><td>SA3</td><td>Difference</td><td>Global Comment #11, 14
Renumbered to align with other similar ICs</td><td>None</td></tr>
<tr><td>Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</td><td>UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown</td><td>Operating Mode Applicability: 1, 2, 3</td><td>Difference</td><td>Global Comment #10</td><td>None</td></tr>
<tr><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.</td><td>(1) a. An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for 15 minutes or longer</td><td>Verbatim</td><td>Global Comment #12</td><td>None</td></tr>
<tr><td>
<table>
<tr><td>[BWR parameter list]</td><td>[PWR parameter list]</td></tr>
<tr><td>Reactor Power</td><td>Reactor Power</td></tr>
<tr><td>RPV Water Level</td><td>RCS Level</td></tr>
<tr><td>RPV Pressure</td><td>RCS Pressure</td></tr>
<tr><td>Primary Containment Pressure</td><td>In-Core/Core Exit Temperature</td></tr>
<tr><td>Suppression Pool Level</td><td>Levels in at least (site-specific number) steam generators</td></tr>
<tr><td>Suppression Pool Temperature</td><td>Steam Generator Auxiliary or Emergency Feed Water Flow</td></tr>
</table>
AND</td><td>Table S-1 Safety System Parameters
• Reactor Power
• RPV Water Level
• RPV Pressure
• Primary Containment Pressure
• Suppression Pool Level
• Suppression Pool Temperature
AND</td><td>Difference</td><td>Global Comment #4, 8</td><td>None</td></tr>
</tbody>
</table>

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA2 (cont.)	b. **ANY** of the following transient events in progress. • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor scram [BWR] / trip [PWR] • ECCS (SI) actuation • Thermal power oscillations greater than 10% [BWR]	b. **Any** of the Table S-2 transient events are in progress **Table S-2 Significant Transients** • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor scram • ECCS actuation • Thermal power oscillations greater than 10%	Difference	Global Comment #4, 9 Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA5	Recognition Category: SA5	SA6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	Automatic or manual scram fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	Difference	Global Comment #4 & 9	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	Global Comment #10 DAEC can be up to 12% power in STARTUP Mode, so Mode 2 applicability added	V33
	(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic or manual scram did not shutdown the reactor.	Difference	Global Comment #4, 9 & 12	None
	AND b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	**AND** b. **ALL** of the following manual actions taken at 1C05 are not successful in lowering reactor power below 5% power • Manual Scram Pushbuttons • Mode Switch to Shutdown • Alternate Rod Insertion (ARI)	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA9	Recognition Category: SA9	SA8	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) a. The occurrence of ANY of the following hazardous events:	(1) a. The occurrence of ANY of the Table S-3 hazardous events:	Difference	Global Comment #12 & 13	None
	• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager	Table S-3 Hazardous Events • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director	Difference	Global Comment #8 & 9	None

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA9 (cont.)	**AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.	**AND** b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode. **AND**	Deviation	Adopted the revised EAL structure and wording provided in approved EAL FAQ 2016-02.	V17
	OR 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	2. **EITHER** of the following: • Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or • The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.	Deviation	Adopted the revised EAL wording provided in approved EAL FAQ 2016-02	V17
			Difference	Added the following clarification to the Basis from EALFAQ 2018-04: An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under SA8 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement. Intent and meaning of the EALs are not altered.	V18

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Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SS1	Recognition Category: SS1	SS1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Loss of **ALL** offsite and **ALL** onsite AC power to essential buses for 15 minutes or longer.	Difference	Global Comment #13, 15	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses for 15 minutes or longer.	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SS5	Recognition Category: SS5	SS6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Inability to shutdown the reactor causing a challenge to (core cooling [PWR] / RPV water level [BWR]) or RCS heat removal.	Inability to shutdown the reactor causing a challenge to RPV water level or RCS heat removal.	Difference	Global Comment #4	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	Global Comment #10 DAEC can be up to 12% power in STARTUP Mode, so Mode 2 applicability added	V33
	(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic or manual scram did not shutdown the reactor.	Difference	Global Comment #4, 9 & 12	None
	AND b. All manual actions to shutdown the reactor have been unsuccessful.	**AND** b. All manual actions to shutdown the reactor have been unsuccessful.	Verbatim		None
	AND c. EITHER of the following conditions exist: • (Site-specific indication of an inability to adequately remove heat from the core) • (Site-specific indication of an inability to adequately remove heat from the RCS)	**AND** c. **EITHER** of the following conditions exist: • RPV level cannot be restored and maintained above -25 inches. **OR** • HCL (Graph 4 of EOP 2) exceeded.	Difference	Global Comment #9	V34 V27
				Intent and meaning of the EALs are not altered.	

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SS8	Recognition Category: SS8	SS2	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Loss of all Vital DC power for 15 minutes or longer.	Loss of **ALL** Vital DC power for 15 minutes or longer.	Difference	Global Comment #13	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) Indicated voltage is less than (site-specific bus voltage value) on **ALL** (site-specific Vital DC busses) for 15 minutes or longer.	(1) Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V12

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SG1	Recognition Category: SG1	SG1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Prolonged loss of all offsite and all onsite AC power to emergency buses.	Prolonged loss of **ALL** offsite and **ALL** onsite AC power to essential buses.	Difference	Global Comment #13, 15	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) a. Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses).	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses	Difference	Global Comment #9 & 13	None
	AND b. EITHER of the following: • Restoration of at least one AC emergency bus in less than (site-specific hours) is not likely.	**AND** b. **EITHER** of the following: • Restoration of at least one AC essential bus in less than 4 hours is not likely. **OR**	Difference	Global Comment #9 & 13	
	• (Site-specific indication of an inability to adequately remove heat from the core)	• RPV level cannot be restored and maintained above -25 inches.	Difference	Global Comment #9	V34
				Intent and meaning of the EALs are not altered.	

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SG8	Recognition Category: SG8	SG2	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Loss of all AC and Vital DC power sources for 15 minutes or longer.	Loss of **ALL** AC and Vital DC power sources for 15 minutes or longer.	Verbatim	Global Comment #13	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses for 15 minutes or longer.	Difference	Global Comment #9, 12, 13	None
	AND b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer.	**AND** b. Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.	Difference	Global Comment #9 & 13 Intent and meaning of the EALs are not altered.	V12

APPENDIX A – ACRONYMS AND ABBREVIATIONS

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS	AC.......Alternating Current	AC.......Alternating Current	Verbatim		N/A
	AOP......Abnormal Operating Procedure	AOP......Abnormal Operating Procedure	Verbatim		N/A
	APRM...Average Power Range Meter		Difference	Not used	N/A
	ATWS...Anticipated Transient Without Scram	ATWS...Anticipated Transient Without Scram	Verbatim		N/A
	B&W....Babcock and Wilcox		Difference	Not used	N/A
	BIIT......Boron Injection Initiating Temperature		Difference	Not used	N/A
	BWR....Boiling Water Reactor	BWR....Boiling Water Reactor	Verbatim		N/A
	CDE......Committed Dose Equivalent	CDE......Committed Dose Equivalent	Verbatim		N/A
	CFR......Code of Federal Regulations	CFR......Code of Federal Regulations	Verbatim		N/A
	CTMT/CNMT...Containment		Difference	Not used	N/A
	CSF......Critical Safety Function		Difference	Not used	N/A
	CSFST...Critical Safety Function Status Tree		Difference	Not used	N/A
	DBA......Design Basis Accident		Difference	Not used	N/A
	DC........Direct Current	DC........Direct Current	Verbatim		N/A
	EAL.......Emergency Action Level	EAL.......Emergency Action Level	Verbatim		N/A
	ECCS....Emergency Core Cooling System	ECCS....Emergency Core Cooling System	Verbatim		N/A
	ECL.......Emergency Classification Level	ECL.......Emergency Classification Level	Verbatim		N/A
	EOF......Emergency Operations Facility	EOF......Emergency Operations Facility	Verbatim		N/A
	EOP......Emergency Operating Procedure	EOP......Emergency Operating Procedure	Verbatim		N/A
	EPA......Environmental Protection Agency	EPA......Environmental Protection Agency	Verbatim		N/A
	EPG.....Emergency Procedure Guideline	EPG.....Emergency Procedure Guideline	Verbatim		N/A
	EPIP.....Emergency Planning Implementing Procedure		Difference	Not used	N/A

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	EPR......Evolutionary Power Reactor		Difference	Not used	N/A
	EPRI.....Electric Power Research Institute		Difference	Not used	N/A
	ERG.....Emergency Response Guideline		Difference	Not used	N/A
	FEMA...Federal Emergency Management Agency	FEMA...Federal Emergency Management Agency	Verbatim		N/A
	FSAR....Final Safety Analysis Report		Difference	Not used	N/A
	GE........General Emergency	GE........General Emergency	Verbatim		N/A
	HCTL....Heat Capacity Temperature Limit	HCL....Heat Capacity Limit	Difference	Updated to reflect DAEC EOPs	N/A
	HPCI.....High Pressure Coolant Injection	HPCI.....High Pressure Coolant Injection	Verbatim		N/A
	HSI........Human System Interface		Difference	Not used	N/A
	IC..........Initiating Condition	IC..........Initiating Condition	Verbatim		N/A
	ID.........Inside Diameter	ID.........Inside Diameter	Verbatim		N/A
	IPEEE...Individual Plant Examination of External Events (Generic Letter 88-20)		Difference	Not used	N/A
	ISFSI....Independent Spent Fuel Storage Installation	ISFSI....Independent Spent Fuel Storage Installation	Verbatim		N/A
	Keff.....Effective Neutron Multiplication Factor	Keff.....Effective Neutron Multiplication Factor	Verbatim		N/A
	LCO.....Limited Condition of Operation	LCO.....Limited Condition of Operation	Verbatim		N/A
	LOCA...Loss of Coolant Accident	LOCA...Loss of Coolant Accident	Verbatim		N/A
	MCR....Main Control Room		Difference	Not used	N/A
	MSIV...Main Steam Isolation Valve		Difference	Not used	N/A
	MSL......Main Stem Line		Difference	Not used	N/A
	mR, mRem, mrem, mREM....milli-Roentgen Equivalent Man	mR, mRem, mrem, mREM....milli-Roentgen Equivalent Man	Verbatim		N/A
	MW.....Megawatt	MW.....Megawatt	Verbatim		N/A
	NEI.......Nuclear Energy Institute	NEI.......Nuclear Energy Institute	Verbatim		N/A
	NPP......Nuclear Power Plant		Difference	Not used	N/A

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	NRC.....Nuclear Regulatory Agency	NRC.....Nuclear Regulatory Agency	Verbatim		N/A
	NSSS....Nuclear Steam Supply System		Difference	Not used	N/A
	NORAD...North American Aerospace Defense Command	NORAD...North American Aerospace Defense Command			N/A
	(NO)UE...(Notification of) Unusual Event	NOUE...Notification of Unusual Event	Difference	DAEC uses full NOUE terminology	N/A
	NUMARC....Nuclear Management and Resources Council	NUMARC....Nuclear Management and Resources Council	Verbatim		N/A
	OBE.....Operating Basis Earthquake	OBE.....Operating Basis Earthquake	Verbatim		N/A
	OCA.....Owner Controlled Area	OCA.....Owner Controlled Area	Verbatim		N/A
	ODCM/ODAM....Offsite Dose Calculation (Assessment) Manual	ODAM...Offsite Dose Assessment Manual	Difference	DAEC uses ODAM	N/A
	ORO.....Offsite Response Organization		Difference	Not used	N/A
	PA.........Protected Area	PA.........Protected Area	Verbatim		N/A
	PACS....Priority Information and Control System		Difference	Not used	N/A
	PAG......Protective Action Guideline	PAG......Protective Action Guideline	Verbatim		N/A
	PICS.....Process Information and Control System		Difference	Not used	N/A
	PRA/PSA...Probabilistic Risk Assessment/Probabilistic Safety Assessment	PRA/PSA...Probabilistic Risk Assessment/Probabilistic Safety Assessment	Verbatim		N/A
	PWR....Pressurized Water Reactor	PWR....Pressurized Water Reactor	Verbatim		N/A
	PS.........Protection System		Difference	Not used	N/A
	PSIG....Pounds per Square Inch	PSIG....Pounds per Square Inch	Verbatim		N/A
	R..........Roentgen	R..........Roentgen	Verbatim		N/A
	RCC....Reactor Control Console		Difference	Not used	N/A
	RCIC...Reactor Core Isolation Cooling	RCIC...Reactor Core Isolation Cooling	Verbatim		N/A

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	RCS.....Reactor Coolant System	RCS.....Reactor Coolant System	Verbatim		N/A
	Rem, rem, REM...Roentgen Equivalent Man	Rem, rem, REM...Roentgen Equivalent Man	Verbatim		N/A
	RETS....Radiological Effluent Technical Specifications		Difference	Not used	N/A
	RPS......Reactor Protection System	RPS......Reactor Protection System	Verbatim		N/A
	RPV......Reactor Pressure Vessel	RPV......Reactor Pressure Vessel	Verbatim		N/A
	RVLIS...Reactor Vessel Level Instrumentation System		Difference	Not used	N/A
	RWCU...Reactor Water Cleanup	RWCU...Reactor Water Cleanup	Verbatim		N/A
	SAR.......Safety Analysis Report		Difference	Not used	N/A
	SAS........Safety Automation System		Difference	Not used	N/A
	SBO.......Station Blackout		Difference	Not used	N/A
	SCBA.....Self-Contained Breathing Apparatus	SCBA.....Self-Contained Breathing Apparatus	Verbatim		N/A
	SG..........Steam Generator		Difference	Not used	N/A
	SI...........Safety Injection		Difference	Not used	N/A
	SICS......Safety Information Control System		Difference	Not used	N/A
	SPDS.....Safety Parameter Display System	SPDS.....Safety Parameter Display System	Verbatim		N/A
	SRO.......Senior Reactor Operator		Difference	Not used	N/A
	TEDE.....Total Effective Dose Equivalent	TEDE.....Total Effective Dose Equivalent	Verbatim		N/A
	TOAF.....Top of Active Fuel	TAF.....Top of Active Fuel	Difference	Updated to reflect DAEC EOPs	N/A
	TSC........Technical Support System	TSC........Technical Support System	Verbatim		N/A
	-	UFSAR....Final Safety Analysis Report	Difference	Used in Section 3.1	N/A
	WOG.....Westinghouse Owners Group		Difference	Not used	N/A

APPENDIX B - DEFINITIONS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	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.	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.	Verbatim		None
	General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.	General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.	Verbatim		None
	Notification of Unusual Event: Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	Unusual Event: Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	Difference	See Global Comment #3	None

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.	Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.	Verbatim		None
	Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.	Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.	Verbatim		None
	Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: • Notification of Unusual Event (NOUE) • Alert • Site Area Emergency (SAE) • General Emergency (GE)	Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: • Notification of Unusual Event (NOUE) • Alert • Site Area Emergency (SAE) • General Emergency (GE)	Verbatim		None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.	Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.	Verbatim		None
	Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.	Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.	Verbatim		None
	CONFINEMENT BOUNDARY: (Insert a site-specific definition for this term.) **Developer Note** – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.	CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. This corresponds to the pressure boundary for the Dry Shielded Canister (DSC) shell (including the inner bottom cover plate) base metal and associated confinement boundary welds.	Difference	Removed developer notes and added site-specific language.	None
	CONTAINMENT CLOSURE: (Insert a site-specific definition for this term.) **Developer Note** – The procedurally defined conditions or actions taken to secure containment (primary or secondary for BWR) and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.	CONTAINMENT CLOSURE: Site specific procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.	Difference	Removed developer notes and added existing definition from present EALs.	None
		DESIGN BASIS EARTHQUAKE (DBE): A DBE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.	Difference	Added term used in HU2 versus use of footnotes	None

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.	EXPLOSION: A rapid, violent, and catastrophic failure of a piece of equipment due to combustion, chemical reaction, or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.	Verbatim		None
	FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized. Developer Note – This term is applicable to PWRs only.		Difference	Term not used for BWRs	None
	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.	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.	Verbatim		None
	HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.	HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.	Verbatim		None

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
	HOSTILE ACTION: An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).	HOSTILE ACTION: An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).	Difference	Spelled out 'NPP' in 2 places	None
	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.	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.	Verbatim		None
	IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.	IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.	Verbatim		None
	INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.	INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.	Verbatim		None

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	NORMAL LEVELS: As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.		Difference	Term not used in this EAL scheme	None
		OPERATING BASIS EARTHQUAKE (OBE): An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.	Difference	Added term used in HU2 versus use of footnotes	None
	OWNER CONTROLLED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the site property owned by, or otherwise under the control of, the licensee. In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan.	OWNER CONTROLLED AREA: The site property owned by or otherwise under the control of the licensee.	Difference	Definition from developer notes used. Developer Notes deleted.	None
	PROJECTILE: An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.	PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.	Difference	Spelled out 'NPP'	None

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
	PROTECTED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the area under continuous access monitoring and control, and armed protection as described in the site Security Plan.	PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.	Difference	Definition from developer notes used. Developer Notes deleted.	None
APPENDIX B - DEFINITIONS	REFUELING PATHWAY: (Insert a site-specific definition for this term.) **Developer Note** – This description should include all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.	REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool, and fuel transfer canal.	Difference	DAEC-specific definition supplied. Developer Notes deleted.	None
	RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection. **Developer Note** – This term is applicable to PWRs only.		Difference	Not used	None
	SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related. **Developer Note** – This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.	SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.	Difference	Removed developer notes and clarified last sentence.	None
	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	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	Verbatim		None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
	of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.	of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.			
		SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the Company. UFSAR Figure 1.2-1 identifies the DAEC SITE BOUNDARY.	Difference	Defined term from ODCM needed for several EALs	None
	UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.	UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.	Verbatim		None
APPENDIX B - DEFINITIONS	UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.	UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.	Verbatim		N/A
	VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.	VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.	Deviation	Updated to reflect wording and guidance of approved EAL FAQ 2016-02. The updated wording clarifies damage assessment meriting an ALERT declaration as used in ICs using this definition (CA6 and SA9).	V17

APPENDIX C - Permanently Defueled ICs/EALs

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
Appendix C – Permanently Defueled ICs/EALs	Appendix C - Permanently Defueled ICs/EALs	Not used at DAEC	Difference	Not applicable to DAEC	None

ATTACHMENT 4

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO

LICENSE AMENDMENT REQUEST TSCR-166

UPDATED SUPPORTING TECHNICAL INFORMATION

248 pages follow

Table 1.1-1 (page 1 of 1)
MODES

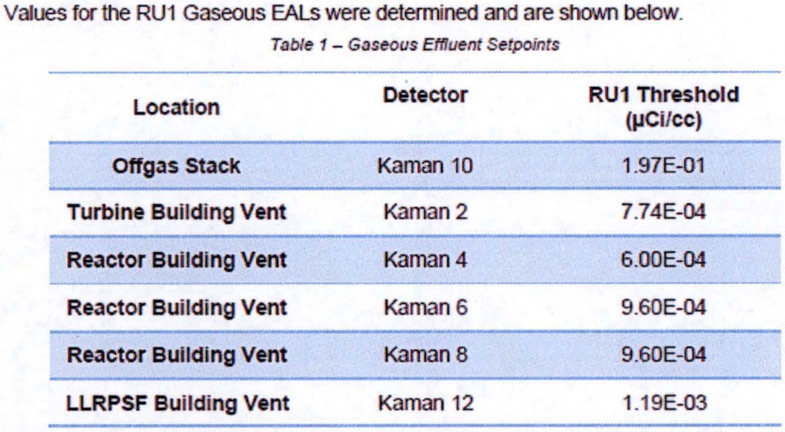
MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel(a) or Startup/Hot Standby	NA
3	Hot Shutdown(a)	Shutdown	> 212
4	Cold Shutdown(a)	Shutdown	≤ 212
5	Refueling(b)	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

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

Development of EAL Threshold values from NEE-323-CALC-003

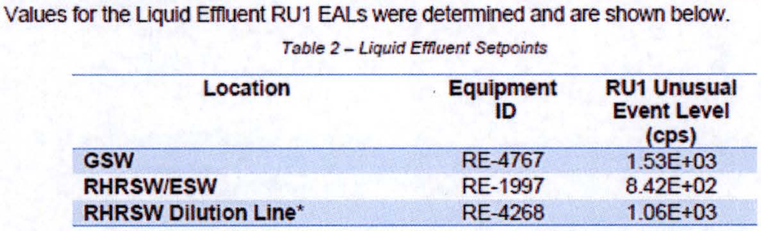
Calculated values are provided in Calc-003 as shown below.



Values for the RU1 Gaseous EALs were determined and are shown below.

Table 1 – Gaseous Effluent Setpoints

Location	Detector	RU1 Threshold (µCi/cc)
Offgas Stack	Kaman 10	1.97E-01
Turbine Building Vent	Kaman 2	7.74E-04
Reactor Building Vent	Kaman 4	6.00E-04
Reactor Building Vent	Kaman 6	9.60E-04
Reactor Building Vent	Kaman 8	9.60E-04
LLRPSF Building Vent	Kaman 12	1.19E-03



Values for the Liquid Effluent RU1 EALs were determined and are shown below.

Table 2 – Liquid Effluent Setpoints

Location	Equipment ID	RU1 Unusual Event Level (cps)
GSW	RE-4767	1.53E+03
RHRSW/ESW	RE-1997	8.42E+02
RHRSW Dilution Line*	RE-4268	1.06E+03

The values are rounded for ease of operator use and to provide a step-wise progression through the emergency classification levels. The resulting values used in the DAEC RU1.1 EAL are shown in the NOUE column below:

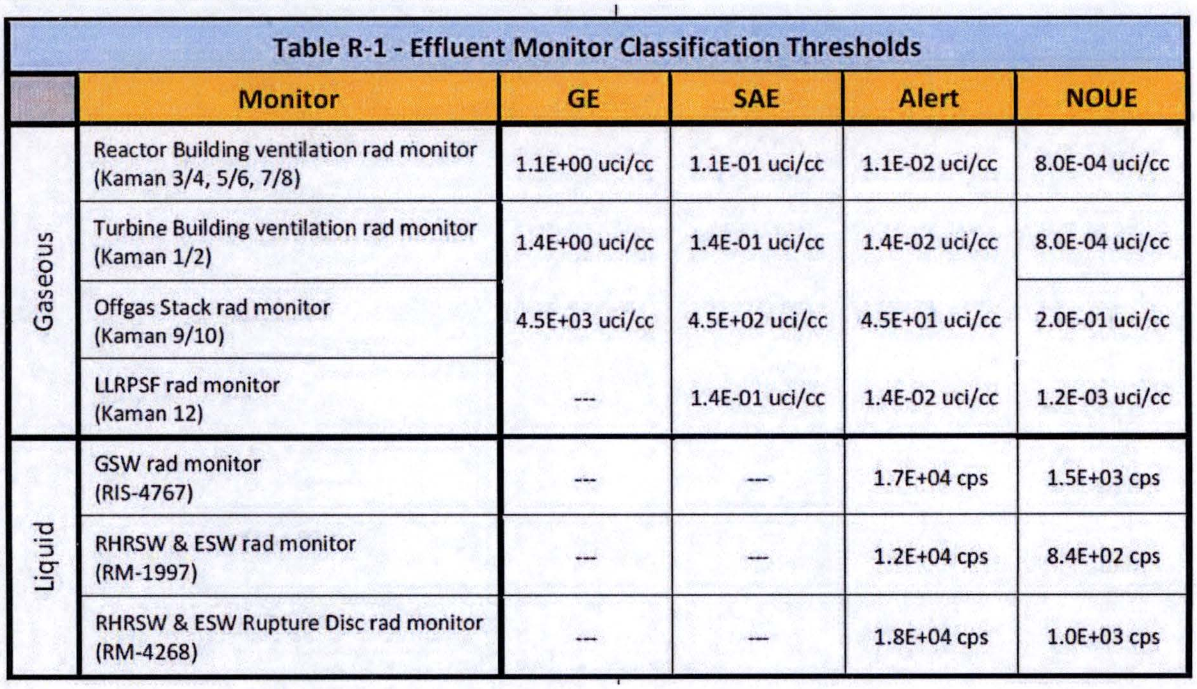


Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

Development of EAL Threshold values from NEE-323-CALC-004

Calculated values are provided in Calc-004 as shown below.

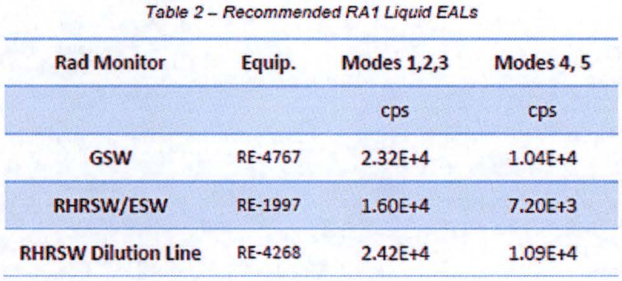
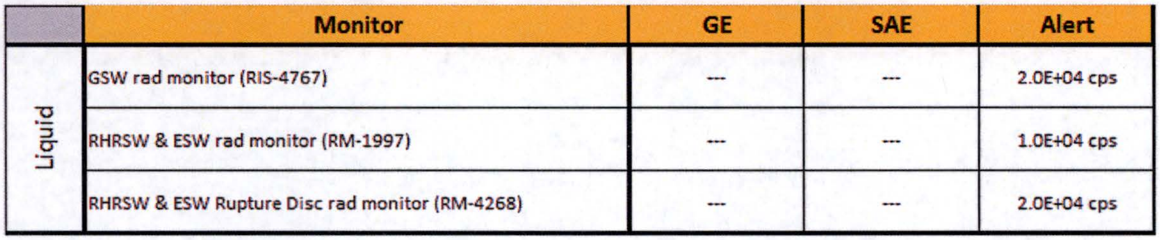


Table 2 – Recommended RA1 Liquid EALs

Rad Monitor	Equip.	Modes 1,2,3	Modes 4, 5
		cps	cps
GSW	RE-4767	2.32E+4	1.04E+4
RHRSW/ESW	RE-1997	1.60E+4	7.20E+3
RHRSW Dilution Line	RE-4268	2.42E+4	1.09E+4

The following table of threshold values was developed for use in the DAEC EAL scheme by averaging the separate Mode 1-3 and Mode 4-5 thresholds from Calc-004, and then rounding the average values for ease of EAL evaluator use, as well as to provide a step-wise progression through the emergency classification.



	Monitor	GE	SAE	Alert
Liquid	GSW rad monitor (RIS-4767)	---	---	2.0E+04 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.0E+04 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	2.0E+04 cps

Development of EAL Threshold values from NEE-323-CALC-002

Due to elevated background radiation levels on these monitors during plant operation (10-12 R/hr), the calculated threshold value was rounded to 5 (minimum serviceable threshold value accounting for scale of monitor) for ease of use by the EAL evaluator, and the "in Mode 5 only" caveat is added to the EAL usage.

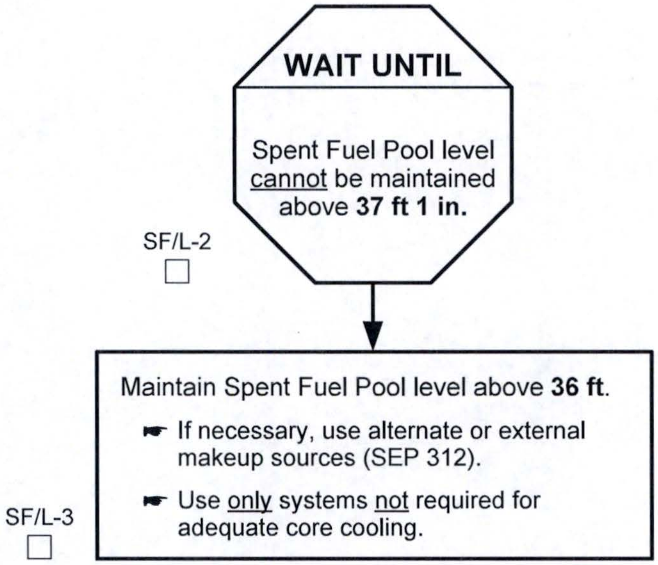
The resultant EALs are:

RA2.2 Reading greater than 5 R/hr on ANY of the following radiation monitors (in Mode 5 only):

- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

CS1/CG1 Core uncovery is indicated by ANY of the following:

- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr



WAIT UNTIL

Spent Fuel Pool level <u>cannot</u> be maintained above **37 ft 1 in.**

SF/L-2 ☐

Maintain Spent Fuel Pool level above **36 ft**.

☛ If necessary, use alternate or external makeup sources (SEP 312).

☛ Use <u>only</u> systems <u>not</u> required for adequate core cooling.

SF/L-3 ☐

DISCUSSION

If spent fuel pool level cannot be restored and maintained above the low level alarm setpoint, an alternate control band is established above the higher of the spent fuel pool level LCO (36 ft.) or the Minimum Safe Operating Spent Fuel Pool Level (25.17 ft.). If necessary, normal spent fuel pool makeup may be augmented by one or more of the alternate and external sources listed in SEP 312.

The Minimum Safe Operating Spent Fuel Pool Level is generically defined to be the lowest water level providing adequate radiation shielding to (1) protect personnel performing local operations required by the EOPs and (2) allow unrestricted access to the main control room. At the DAEC, the Minimum Safe Operating Spent Fuel Pool Level is defined consistent with NEI 12-02 Level 2, described as the level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck. The corresponding spent fuel pool level at the DAEC is defined to be 25.17 ft., approximately 10 ft. above the top of the fuel racks.

Local Operations for Operating and Normal Shutdown/Cooldown

Procedure Section and Step	Step Action	If action not performed, does this prevent shutdown or cooldown?	Building	Elevation	Room	Mode
IPOI 3, Section 5, step (9)	Between 50% and 60% Reactor Power shutdown one Condensate and Reactor Feed Pump per OI 644 unless otherwise directed by CRS.	No. The Feed Pumps and Condensate Pumps can be tripped from the Control Room if necessary, and HPCI and/or RCIC can be used to maintain RPV Level.	N/A	N/A	N/A	N/A
IPOI 3, Section 5, step (10)	When turbine load is lowered to approximately 200 MWe, remove the 1E-18A[B] 2nd Stage Reheat System from service in accordance with OI 646, Extraction Steam.	No. 2nd Stage Reheat can be left in service and the turbine can be tripped if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (10)	Secure condensate demineralizers as directed by OI 639, Section 5.1.	No. Condensate Demineralizers will automatically go into the "hold" mode as power and flow are lowered.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (11)	Commence primary containment purge per OI 573.	No. This is only necessary if a Drywell entry is anticipated.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (13)	At the refueling bridge, verify that the Main Disconnect is closed and that the SYSTEM START pushbutton has been depressed.	No. Control rod insertion will not be inhibited.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (14)	Prior to disconnecting the generator from the grid, perform the following: (a) If needed, start up the Auxiliary Boiler per OI 727.	No. Aux Boiler is not required to accomplish shutdown.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (22)	Following Turbine Trip: (a) Verify that Reactor Coolant Chloride and Conductivity analyses have been performed. (b) Operate the Turbine Lube Oil and Turning Gear System per OI 693.3. (c) Shut down the generator per OI 698. (d) Shut down the turbine per OI 693.1.	No. These systems can be left in service if necessary.	N/A	N/A	N/A	N/A

Procedure Section and Step	Step Action	If action not performed, does this prevent shutdown or cooldown?	Building	Elevation	Room	Mode
IPOI 4, Section 3 step (24)	Shut down the following generator support systems, as desired: Isolated Phase Bus Cooling - OI 698, Stator Water Cooling - OI 697, H_2 Seal Oil - OI 695.1, H_2 and CO_2 Gas - OI 695.2	No. These systems can be left in service if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (26)	Secure hydrogen, oxygen and/or air injection per OI 563, Hydrogen Water Chemistry.	No. The Hydrogen Water Chemistry System will secure itself if left in service.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (27)	As directed by the CRS, perform the following steps as necessary to limit reactor vessel depressurization following the reactor scram: (b) Start 1P32 Mechanical Vacuum Pump per OI 691. (c) Secure the SJAEs and Offgas per OI 691 and OI 672.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (6)	For the remainder of this section use the following methods as necessary to cooldown and depressurize the reactor vessel to maintain a controlled cooldown rate less than the TS Limit of 100°F in any 1 hour period. (a) Use the Main Turbine Bypass Valve to control cooldown per OI 693.1 Section 4.5 if available, (b) If desired cooldown with RCIC per OI 150 (preferred method if MSIVs are closed), (c) If desired cooldown with HPCI per OI 152 (RCIC may become inadequate as pressure lowers) (d) Control steam flow from the reactor vessel to the main condenser through steam seals and steam drains, (e) Secure steam seals per OI 692 as required to limit cooldown after the turbine is on the jack and vacuum is broken.	(a) No. The MSIVs can be closed if necessary to limit plant cooldown rate. (b) No – operated from the Control Room (c) No – Operated from the Control Room (d) No. The MSIVs can be closed if necessary to limit plant cooldown rate. (e) No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A

Procedure Section and Step	Step Action	If action not performed, does this prevent shutdown or cooldown?	Building	Elevation	Room	Mode
IPOI 4, Section 4 step (7)	As plant cooldown continues perform the following: (NA if MSIVs are closed) (a) Control steam seal pressure 3 to 4 psig using MO-1169, MAIN STEAM SUPPLY, MO-1170, REGULATOR BYPASS and/or MO-1171, MANUAL UNLOADER on 1C07, (b) Start 1P-32 MECHANICAL VACUUM PUMP per OI 691, (c) When reactor pressure approaches 500 psig or cooldown rate cannot be controlled within the limit, then secure SJAEs and Offgas System per OI 691 and OI 672, respectively, if not previously secured, (d) If not using EHC Pressure Set to control plant cooldown, then at 1C07, use the PRESSURE SET ADJUST pushbuttons to maintain A[B] PRESSURE SET DEMAND between 150 and 50 psig above reactor pressure as reactor pressure decreases. Otherwise, N/A.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (8)	At approximately 400 psig, secure the operating feed pump per OI 644.	No. The Feed Pumps and Condensate Pumps can be tripped from the Control Room if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (9)	When RHR Shutdown Cooling Isolation Interlocks can be reset (approximately 100 psig), reset the isolation, then initiate Shutdown Cooling per OI 149.	No, this system can be placed in service from the Control Room if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (10)	Perform the following after the turbine trip, if needed: (a) Verify that Reactor Coolant Chloride and Conductivity analysis has been performed, (b) Operate the Turbine Lube Oil and Turning Gear System per OI 693.3, (c) Shutdown the Main Generator per OI 698, (d) Shutdown the Main Turbine per OI 693.1.	No. These systems can be left in service if necessary.	N/A	N/A	N/A	N/A

Procedure Section and Step	Step Action	If action not performed, does this prevent shutdown or cooldown?	Building	Elevation	Room	Mode
IPOI 4, Section 4 step (11)	Shutdown the following systems as directed by the CRS/OSM. (a) Isolated Phase Bus Cooling per OI 698, (b) Stator Water Cooling per OI 697, (c) H_2 Seal Oil per OI 695.1, (d) H_2 and CO_2 Gas per OI 695.2, (e) Secure SJAEs per OI 691 and Offgas per OI 672 if not previously performed.	No. These systems can be left in service if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (12)	Perform the following at approximately 50 psig: (a) Close the BYPASS VALVE OPENING JACK SELECTOR, (b) Line up and place RFP Stuffing Box Pump 1P-134 in operation to maintain Seal Water Drain Tank 1T-135 level.	No. The Feed Pumps and Condensate Pumps can be tripped from the Control Room if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (13)	When steam seal pressure cannot be maintained or the turbine shaft has cooled per OI 693.3, open Condenser Vacuum Breaker valves V-03-67 and V-03-73.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (14)	Secure MECHANICAL VACUUM PUMP 1P-32 when no longer required per OI 691.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (15)	When the condenser is at atmospheric pressure, secure the Turbine Steam Seal System per OI 692.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (18)	Shut down the operating condensate pump per OI 644 when no longer required for RPV Level Control or Hotwell cleanup recirculation.	No. The Feed Pumps and Condensate Pumps can be tripped from the Control Room if necessary.	N/A	N/A	N/A	N/A

Conclusion of manual action evaluation for EALs RA3 and HA5 is shown below:

EALs RA3 and HA5 are not applicable to DAEC because the evaluation has shown that there are no rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. All areas outside the Control Room that contain equipment necessary for normal plant operation, cooldown and shutdown do not require physical access to operate.

Development of EAL Threshold values from NEE-323-CALC-005

Calculated values are provided in Calc-005 as shown below.

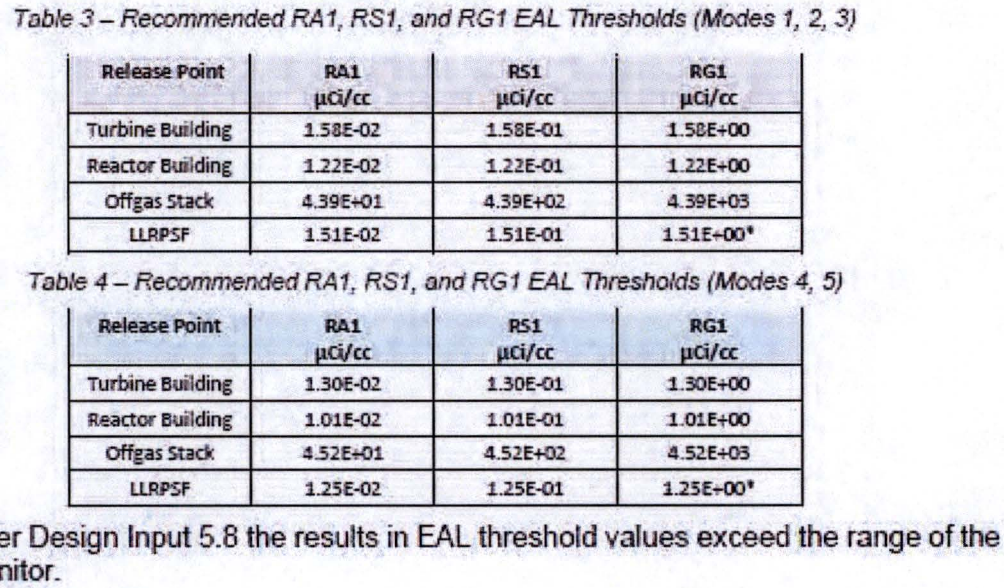


Table 3 – Recommended RA1, RS1, and RG1 EAL Thresholds (Modes 1, 2, 3)

Release Point	RA1 µCi/cc	RS1 µCi/cc	RG1 µCi/cc
Turbine Building	1.58E-02	1.58E-01	1.58E+00
Reactor Building	1.22E-02	1.22E-01	1.22E+00
Offgas Stack	4.39E+01	4.39E+02	4.39E+03
LLRPSF	1.51E-02	1.51E-01	1.51E+00*

Table 4 – Recommended RA1, RS1, and RG1 EAL Thresholds (Modes 4, 5)

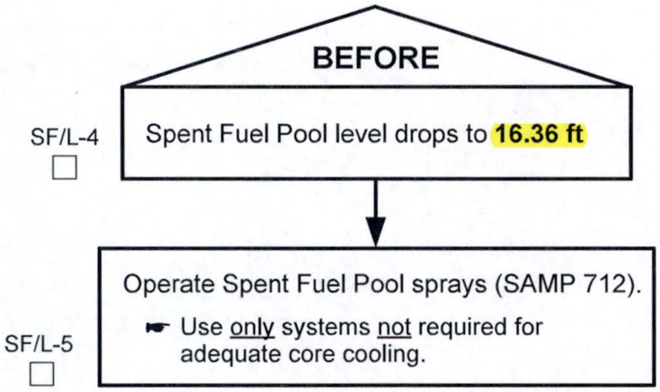
Release Point	RA1 µCi/cc	RS1 µCi/cc	RG1 µCi/cc
Turbine Building	1.30E-02	1.30E-01	1.30E+00
Reactor Building	1.01E-02	1.01E-01	1.01E+00
Offgas Stack	4.52E+01	4.52E+02	4.52E+03
LLRPSF	1.25E-02	1.25E-01	1.25E+00*

* Per Design Input 5.8 the results in EAL threshold values exceed the range of the monitor.

The following table of threshold values was developed for use in the DAEC EAL scheme by averaging the separate Mode 1-3 and Mode 4-5 thresholds from Calc-005, and then rounding the average values for ease of EAL evaluator use, as well as to provide a step-wise progression through the emergency classification. Resulting values are shown in the Alert, SAE, and GE columns below:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps



DISCUSSION

If spent fuel pool level cannot be controlled using alternate or external makeup sources, sprays are used to add water to the spent fuel pool, cool exposed bundles, and reduce radioactivity releases. However, spray operation may damage electrical equipment and flood lower elevations of the secondary containment, complicating implementation of other emergency response strategies, and runoff from sprays could spread radioactivity release. Use of sprays is therefore delayed until it is determined that spent fuel pool level cannot be maintained above the top of the fuel racks. As long as the spent fuel assemblies are covered with water , the fuel will not overheat and efforts should focus on providing sufficient makeup flow to keep the assemblies submerged.

The lowest measurable spent fuel pool level using the wide range instrument is 16.16 ft., approximately one foot above the top of the spent fuel racks. The action level in SF/L-4 corresponds to NEI 12-02 Level 3, the level at which fuel remains covered but actions to implement make-up water addition should no longer be deferred. The "before" condition permits appropriate anticipatory action based on the spent fuel pool leakage rate, radiation levels, available resources, and the time required to place sprays in service. Steps to prepare spray equipment for use should be initiated while radiation levels permit access to the refueling floor and timed to optimize use of available resources.

As in Steps SF/T-3 and SF/L-3, available spray sources may be alternated between RPV injection and spent fuel pool spray modes as long as adequate core cooling can be maintained, but maintaining adequate core cooling takes precedence over spent fuel pool cooling (refer to the discussions of Steps SF/T-3 and SF/L-3 above).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources — Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred and alternate preferred), and the onsite standby power sources (Diesel | Generators (DGs) 1G-31 and 1G-21). As discussed in UFSAR Section 3.1.2.2.8 (Ref. 1), the design of the AC Electrical Power System provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) Systems via essential buses 1A3 and 1A4.

The Class 1E AC Distribution System is divided into redundant load groups, so loss of any one group does not prevent the minimum safety functions from being performed. Each load group has connections to two preferred offsite power supplies and a single DG.

Offsite power is supplied to the 161 kV and 345 kV switchyards from the transmission network by six transmission lines. The 345 kV switchyard and the 161 kV switchyard are connected via the autotransformer, and both sections of the switchyard are connected to the transmission grid by at least two independent lines. From the 161 kV switchyard (the preferred power source), a single overhead transmission line feeds the startup transformer. From the startup transformer, dual isolated secondary windings provide feeds to the 4160 volt essential buses, 1A3 and 1A4, through separate bus supply lines and circuit breakers. The startup transformer is sized to supply all plant power (both essential and non-essential loads) during unit startup. From the tertiary winding on the autotransformer (the alternate preferred power source), a single 34.5 kV underground line feeds the standby transformer. From the standby transformer, a single 4160 volt line feeds both essential buses through separate bus supply circuit breakers. A detailed description of the offsite power network and circuits to the onsite Class 1E essential buses is found in the UFSAR, Sections 8.2.1.3 and 8.3.1.1.5 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls

(continued)

BACKGROUND (continued)

required to transmit power from the offsite transmission network to the onsite Class 1E essential bus or buses. Startup transformer (1X3) provides the normal source of power to the essential buses 1A3 and 1A4. If either 4.16 kV essential bus loses power, an automatic transfer from the startup transformer to the standby transformer (1X4) occurs.

The startup transformer and standby transformer are both sized to accommodate the starting of all ESF loads on receipt of an accident signal. Emergency loads are sequenced onto the essential buses regardless of the source of power (onsite or offsite).

The onsite standby power source for 4.16 kV essential buses 1A3 and 1A4 consists of two DGs. DGs 1G-31 and 1G-21 are dedicated to essential buses 1A3 and 1A4, respectively. A DG starts automatically on a Loss of Coolant Accident (LOCA) signal (i.e., low reactor water level signal or high drywell pressure signal) or on an essential bus degraded voltage or undervoltage signal. After the DG has started, it automatically ties to its respective bus after offsite power is tripped as a consequence of essential bus undervoltage or degraded voltage, independent of or coincident with a LOCA signal. The DGs also start and operate in the standby mode without tying to the essential bus on a LOCA signal alone. Following the trip of offsite power, non emergency loads powered from essential buses are load shed. When the DG is tied to the essential bus, loads are then sequentially connected to its respective essential bus. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG.

In the event of a loss of both the preferred power source and the alternate preferred power source, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a LOCA.

Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading of the DGs in the process. Within 25 seconds after the initiating signal is received, all automatic and permanently

(continued)

ANNUNCIATOR RESPONSE PROCEDURE

ARP 1C08A

GENERATOR AND AUXILIARY POWER

Usage Level
Reference Use

Record the following: Date/Time: ____________ / _________ Initials: _______

NOTE: *User shall perform and document a Temp Issue/Rev. Check to ensure revision is current, in accordance with procedure use and adherence requirements.*

Prepared By: ____________________ / ____________________ Date: __________
Print Signature

CROSS-DISCIPLINE REVIEW (AS REQUIRED)		
Reviewed By: ____________________ Print	/ ____________________ Signature	Date: __________
Reviewed By: ____________________ Print	/ ____________________ Signature	Date: __________

PROCEDURE APPROVAL		
Approved By ____________________ Print	/ ____________________ Signature	Date: __________

ALARM WINDOW ENGRAVINGS AND GRID LAYOUT

1C08A

Same on annunciator panel 1C08B for Division 2

	1	2	3	4	5	6	7	8	9	10	11	12
A	AUX XFMR TO 1A1 BREAKER 1A101 TRIP	BUS 1A1 LOCKOUT TRIP OR LOSS OF VOLTAGE	S/U XFMR TO 1A1 BREAKER 1A102 TRIP	STBY XFMR TO 1A3 BREAKER 1A301 TRIP	BUS 1A3 LOCKOUT TRIP	S/U XFMR TO 1A3 BREAKER 1A302 TRIP	STARTUP XFMR 1X3 TROUBLE	UNINTERRUPTIBLE AC 1Y23 UNDERVOLTAGE OR INVERTER TROUBLE	125 VDC SYSTEM 1 TROUBLE	"A" DIESEL GEN 1G-31 RUNNING	A DG TO BUS 1A3 BREAKER 1A311 TRIP	"A" DIESEL GEN 1G-31 LOCKOUT TRIP
B	1A1 TO XFMR 1X11 BREAKER 1A107 TRIP	1A1 TO XFMR 1X71 BREAKER 1A108 TRIP	1A1 TO XFMR 1X51 BREAKER 1A109 TRIP	SWITCHYARD SUPPLY BREAKER 1A110 TRIP	LC XFMR 1X31 BREAKER 1A303 TRIP	LC XFMR 1X91 BREAKER 1A312 OR MCC 1B91 BKR 1B903 TRIP	MAIN GENERATOR IMPROPER PHASE SEQUENCE	INSTRUMENT AC 1Y21 UNDERVOLTAGE OR INVERTER TROUBLE	125 VDC CHARGER 1D12 TROUBLE	"A" DIESEL GEN FUEL OIL DAY TANK 1T-37A LO-LO-LEVEL	"A" DIESEL GEN 1G-31 PHASE OVERCURRENT OR GROUND FAULT	"A" DIESEL GEN 1G-31 OVERSPEED TRIP
C	XFMR 1X11 TO LC1B1 BREAKER 1B101 TRIP	LC 1B1/1B2 CROSS TIE BREAKER 1B107 TRIP	XFMR 1X51 TO LC 1B5 BREAKER 1B501 TRIP	125 VDC SYSTEM 1 BATTERY 1D1 DISCONNECTED	LC 1B3 BREAKER 1B301, 1B302 1B303 OR 1B304 TRIP	BUS 1A3 LOSS OF VOLTAGE	STARTUP XFMR LOCKOUT TRIP	INSTRUMENT AC 1Y11 UNDERVOLTAGE OR INVERTER TROUBLE	125 VDC CHARGER 1D120 TROUBLE	AUX BOILER FUEL TANK 1T-34 LO LEVEL	"A" DIESEL GEN PANEL 1C-93 TROUBLE	"A" DIESEL GEN 1G-31 ENGINE CRANKING
D	LC 1B1 BREAKER 1B102, 1B103 1B104 OR 1B105 TRIP	LC 1B5/1B6 CROSSTIE BREAKER 1B505 TRIP	LOAD CENTER 1B5 BREAKER 1B502 1B503 OR 1B504 TRIP		MCC 1B34A TIE BKR 1B3401 TRIP	MCC 1B34A/1B44A TIE BREAKER 1B3402 OR 1B4402 TRIP	4KV BUS AUTO TRANSFER INOP	DIESEL FUEL OIL STORAGE TANK 1T-35 LO LEVEL	"A" DIESEL GEN 1G-31 CONTROL POWER FAILURE	"A" DIESEL GEN 1G-31 AUTO START INHIBITED	"A" DIESEL GEN 1G-31 ENGINE SHUTDOWN	"A" DIESEL GEN 1G-31 START FAILURE

ANNUNCIATOR PANEL: 1C08A
COORDINATES: A-9
PAGE: 1 of 3

125V DC
SYSTEM 1
TROUBLE

TITLE: 125 VDC SYSTEM 1 TROUBLE (GROUND OR LOW VOLTAGE)

1.0	PROBABLE CAUSE(S) /	INITIATING DEVICE(S) /	SETPOINT(S)
1.1	Ground fault on 125 VDC System 1	Positive to ground Relay 64 or negative to ground	9 Volt differential Relay 64
1.2	125 VDC System 1 low voltage	Relay 1D10-27	105 VDC (dec)
1.3	Positive or negative metering fuse blown	FU 3 amp fuse blown	blown

2.0 AUTOMATIC ACTIONS

2.1 If due to a complete loss of 125 VDC System 1:

a. Various control systems half trip.

b. Scoop Tube for Recirc Pump A locks up.

c. Breaker 1B3401 auto trips open after 6 second time delay, 1B4401 auto closes.

d. Static Switch JS1501 transfers from Inverter 1D15 to Regulating Transformer 1Y1A.

2.2 If the 125 VDC System 1 is not lost, no AUTOMATIC ACTIONS occur.

ANNUNCIATOR PANEL:	**1C08A**
COORDINATES:	**B-9**
PAGE: **1** of **3**	

125V DC
CHARGER 1D12
TROUBLE

TITLE: 125 VDC CHARGER 1D12 TROUBLE

NOTE

This is a normal alarm anytime Charger 1D12 is being changed over to Charger 1D120.

1.0	PROBABLE CAUSE(S) /	INITIATING DEVICE(S) /	SETPOINT(S)
1.1	AC Breaker 1D12-01 open	Relay K-8	OPEN position
1.2	DC Breaker 1D12-02 open	Relay K-7	OPEN position
	AND		
	DC Breaker 1D12-03 open	Relay K-7	OPEN position
1.3	Charger Failure	Relay K-5	< 5 Amps (dec) 40 second time delay
1.4	Reverse Current	Relay K-4	Reverse Current Detected
1.5	DC Undervoltage	Relay K-2	105 VDC (dec)
1.6	AC Undervoltage	Relay K-1	340 VAC (dec)

2.0 AUTOMATIC ACTIONS

2.1 If due to CAUSE 1.3, 1.4, or 1.5, Charger 1D12 front panel trouble light illuminates.

ANNUNCIATOR PANEL: **1C08A**
COORDINATES: **C-9**
PAGE: **1** of **3**

125V DC
CHARGER 1D120
TROUBLE

TITLE: 125 VDC CHARGER 1D120 TROUBLE

NOTE

This is a normal alarm anytime 1D120 is being changed over to Chargers 1D12 or 1D22.

1.0 PROBABLE CAUSE(S) / INITIATING DEVICE(S) / SETPOINT(S)

	PROBABLE CAUSE(S)	INITIATING DEVICE(S)	SETPOINT(S)
1.1	AC Breaker 1D120-01 open	Relay K-8	Open Position
1.2	DC Breaker 1D120-02 open	Relay K-7	OPEN Position
1.3	DC Breaker 1D120-03 open	Relay K-7	OPEN Position
1.4	Charger Failure	Relay K5	< 5 Amps (dec) 40 Second TD
1.5	Reverse Current	Relay K-4	Reverse Current Detected
1.6	DC Undervoltage	Relay K-2	105 VDC (dec)
1.7	AC Undervoltage	Relay K-1	340 VAC (dec)

2.0 AUTOMATIC ACTIONS

2.1 If due to CAUSE 1.4, 1.5, or 1.6, Charger 1D120 front panel trouble light illuminates.

Table B 3.8.7-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	DIVISION 1[(a)]	DIVISION 2[(a)]
AC safety buses	4160 V	Essential Bus 1A3	Essential Bus 1A4
	480 V	Load Centers 1B3, 1B9	Load Centers 1B4, 1B20
	480 V	Motor Control Centers 1B32, 1B34	Motor Control Centers 1B42, 1B44
125 VDC buses	125 V	Distribution Panels 1D10, 1D11, 1D13 RCIC Motor Control Center 1D14	Distribution Panels 1D20, 1D21, 1D23
250 VDC buses	250 V	N/A	Distribution Panel 1D40 Motor Control Centers 1D41 and 1D42

(a) Each division of the AC and DC electrical power distribution systems is a subsystem.

ABNORMAL OPERATING PROCEDURE

AOP 302.1

LOSS OF 125 VDC POWER

Usage Level
Reference Use

NOTE

This AOP is normally coordinated by the Reactor Operator.

Record the following: Date/Time: ____________ / ________ Initials: _______

NOTE: *User shall perform and document a Temp Issue/Rev. Check to ensure revision is current, in accordance with procedure use and adherence requirements.*

Enter the following as applicable:

LOSS OF 125 VDC DIV I

IMMEDIATE ACTIONS

1. Place RX WATER LEVEL CONTROL INPUT SELECT HSS-4560 in A LEVEL at 1C05. ________

2. Place 1 OR 3 ELEMENT CONTROL SELECT HSS 4450 in 1 ELEMENT at 1C05. ________

AUTOMATIC ACTIONS

- 1G001 Voltage Regulator transfers to manual with no adjustment capability
- Loss of 1A1/1A3 breaker control
- MG Set A Scoop Tube Power Failure Lockout initiates (no Amber lockup light)
- SBGTS A SV-5801A fails closed, SV-5815A, SV-5817A and SV-5825A fail open
- MCC 1B34A/1B44A will auto transfer to 1B44
- JS1501 will transfer from Inverter 1D15 to Reg Trans 1Y1A
- Group 3A Primary Containment Isolation (Lockout relay will not trip)
- Rx FEEDWATER FLOW B FEEDLINE FI-1626 fails downscale
- STEAM FLOW B STEAMLINE FI-4409 fails downscale
- Recirc Lube Oil Pumps 1P202A & 1P202B trip, and 1P202C auto starts on low oil pressure (continued operation of the A Recirc MG Set is allowable in this condition).
- B GEMAC LEVEL LI4560 fails low
- HPCI will not trip on high level
- Loss of 1D10 with B LEVEL selected and no operator action results in:
  - Feedwater control opens Feed Reg Valves
  - Reactor water level goes high
  - "B" Reactor Feed Pump and Main Turbine Trip on high level ("A" Reactor Feed Pump cannot be tripped remotely)
  - Reactor Scram (Main turbine trip)
  - Loss of 1A1 and 1A2 power (failure to auto transfer)
  - If >26% power, Reactor Recirc. Pumps 1P-201A and 1P-201B trip (RPT)

LOSS OF 125 VDC DIV I

FOLLOW-UP ACTIONS

NOTE

Follow up actions may be performed in any order.

1. Establish critical parameter monitoring of RPV Water Level, as priorities allow.
2. Stabilize reactor power level and maintain recirculation loop flows balanced. Use manual control of the A MG Set and Recirc Pump from the MG Set Room in accordance with OI 264.

NOTE

Buses 1A1 and 1A2 will not auto transfer to the startup transformer on a turbine trip.

3. Transfer Bus 1A2 to the startup transformer per OI 304.1.
4. **IF** 1A4 has power available **THEN** verify TIE BREAKER 1B4401 MCC 1B44A/1B34A is closed.
5. **IF** a Reactor Scram occurs: **THEN** perform the following:
   a. Verify main turbine trip.
   b. Verify the H and I breakers open.
   c. Only after H and I are open, direct an operator to trip the GENERATOR EXCITER FIELD BREAKER locally.
6. Reference EPIP 1.1 for EAL assessment, for instrumentation, perform alarm panel checks as needed to confirm threshold is met.
7. Suspend all evolutions in progress associated with electrical switchgear and switching operations.
8. Locally operate affected switchgear to start and stop equipment as required.

FOLLOW-UP ACTIONS (continued)

NOTE

Loss of 125 VDC DIV 1 causes a loss of 1A1 control power. If 1A1 is on the aux transformer, all 1A1 loads will remain energized with no automatic trips or starts until the Main Generator is tripped. Likewise, when the main generator is tripped, All 1A1 loads will be lost until manually restored.

1A1 loads:

A Feed Pump	(1A103) – no 211" trip
A Condensate Pump	(1A106)
A Recirc MG Set	(1A104)
A Circ Water Pump	(1A105)
Load Center 1B1	(1A107)
Load Center 1B5	(1A109)
Load Center 1B7	(1A108)

9. **IF** 1A1 is deenergized **THEN** reenergize 1A1 locally:

   a. Trip 1A101.

   b. Strip 1A1 loads.

   c. Close 1A102.

   d. Restart loads as required.

NOTE

If operation of the RCIC System is required while the Division I 125 VDC System is deenergized, use the HPCI System in manual mode.

10. Evaluate the status of the 125 VDC Electrical Distribution System to determine the cause of the malfunction.

LOSS OF 125 VDC DIV I

FOLLOW-UP ACTIONS (continued)

11. **IF** 1D10 is totally deenergized **THEN** perform the following:

    a. Open all branch circuit breakers at Panels 1D10, 1D11, 1D13, and 1D14. ________

    b. Verify MCC 1B32 energized. ________

    c. Locally inspect circuit breakers at Panels 1D10, 1D11, 1D13, and 1D14. ________

    d. Verify Battery Room ventilation. ________

    e. Locally inspect Battery 1D1 ________

    f. Locally inspect Battery Chargers 1D12 and 1D120. ________

    g. Restore power to 1D10 and branch panels. ________

    h. Comply with Tech Spec Requirements for “Distribution Systems - Operating” or “Distribution Systems - Shutdown”, as applicable. ________

    i. Comply with ACP 1412.4 Impairments to Fire Protection System. ________

12. **IF** 1D10 is not lost, but one or more loads are lost **THEN** investigate and evaluate the status of the individually affected loads and comply with Tech Spec Requirements for “Distribution Systems - Operating” or “Distribution Systems - Shutdown”, as applicable. ________

13. **IF** 125 VDC Div I Battery 1D1 is made or found to be inoperable **THEN** comply with Tech Spec Requirements for “DC Sources - Operating” or “DC Sources - Shutdown”, as applicable. ________

14. **IF** 125 VDC Div I Battery 1D1 is < 110 VDC **THEN** Verify battery cell parameters per STP 3.8.4-02 within 24 hours. ________

15. **WHEN** Div I 125 VDC System is restored and the use of-TIE BREAKER 1B3401 is desired **THEN** send an operator to TIE BREAKER 1B3401 MCC 1B34A/44A to press reset button prior to closing. ________

16. **WHEN** power is restored **THEN** reset A Scoop Tube lockout per OI 264. ________

LOSS OF 125 VDC DIV I

PROBABLE ANNUNCIATORS

1C08A,	A-2	Bus 1A1 LOCKOUT TRIP OR LOSS OF VOLTAGE
	A-7	STARTUP XFMR 1X3 TROUBLE
	A-9	125 VDC SYSTEM 1 TROUBLE
	B-9	125 VDC CHARGER 1D12 TROUBLE
	C-4	125 VDC SYSTEM 1 BATTERY 1D1 DISCONNECTED
	C-6	BUS 1A3 LOSS OF VOLTAGE
	C-8	INSTRUMENT AC 1Y11 UNDERVOLTAGE OR INVERTER TROUBLE
	C-9	125 VDC CHARGER 1D120 TROUBLE
	D-5	MCC 1B34A TIE BKR 1B3401 TRIP
	D-9	A DIESEL GEN 1G-31 CONTROL POWER FAILURE
1C08C,	A-3	MAIN GENERATOR LOCKOUT RELAY CKT LOSS OF 125 VDC
	B-3	MAIN GENERATOR VOLTAGE REGULATION IN MANUAL
	B-6	H_2/STATOR COOLING PANEL 1C83 LOSS OF DC

PROBABLE INDICATIONS

Annunciators - Loss of power to the following panels:

1C03
1C04
1C14
1C05
1C08B
1C09A
1C23A
1C24A
1C22
1C25A
1C26A
1C34
1C35A
1C40
1C40A

1C08 - Loss of the following:

- 4160V BUS 1A1 switchgear control, indication and automatic trip functions
- 480V LC 1B1 switchgear control and indication
- 480V LC 1B7 switchgear control and indication
- 480V LC 1B5 switchgear control and indication
- 4160V BUS 1A3 switchgear control, indication and automatic trip functions
- 480V LC 1B3 switchgear control and indication
- 480V LC 1B9 switchgear control and indication
- TIE BREAKER 1B3401 MCC 1B34A/1B44A control and indication
- GENERATOR EXCITER FIELD BREAKER control and indication

1C07 - Loss of the following:

- EMERGENCY BEARING OIL PUMP 1P-40 control and indication

PROBABLE INDICATIONS (continued)

1C06 - Loss of the following:

- A CIRC WATER PUMP 1P-4A control and indication
- A CONDENSATE PUMP 1P-8A control and indication
- A REACTOR FEEDWATER PUMP 1P-1A control and indication
- A GSW PUMP 1P-89A control and indication
- A[C] RWS PUMP 1P-117A and C control and indication

1C05 - Loss of the following:

- A WIDE RANGE LEVEL LI-4539 indication fails low
- B GEMAC LEVEL LI-4560 fails low
- B REACTOR PRESS PI-4564 fails low
- A CRD PUMP 1P-209A control and indication
- Rx FEEDWATER FLOW B FEEDLINE FI-1626 fails downscale
- STEAM FLOW B STEAMLINE FI-4409 fails downscale

1C04 - Loss of the following:

- MG SET LUBE OIL PUMP 1P-202A control and indication
- MG SET LUBE OIL PUMP 1P-202B control and indication
- A MG SET SPEED CONTROL control and indication
- A MG SET EMERG AUX OIL PUMP 1P-204A control and indication
- RCIC TURBINE CONTROL VALVE HV-2406 position indication
- RCIC System Drain Valves RCIC STEAM LINE DRAIN ISOL CV-2410, and CLOSED RADWASTE DISCH ISOL CV-2435
- RCIC System Condensate Pump Motor and Motor Operated Valves control and indication
- Inboard MSIVs indication and DC Solenoid Valve control

1C03 - Loss of the following:

- A CORE SPRAY PUMP 1P-211A control and indication
- A RHR PUMP 1P-229A and C RHR PUMP 1P-229C control and indication
- A RHRSW PUMP 1P-22A and C RHRSW PUMP 1P-22C control and indication
- A RHR HX SHELL OUTBD VENT MO-2044A and A RHR HX SHELL INBD VENT MO-2044B percent indication
- HPCI STEAM LINE DRAIN ISOL CV-2211 and CLOSED RADWASTE DISCH ISOL CV-2234 control and indication

LOSS OF 125 VDC DIV I

INFORMATION

- 1D10 125 VDC Distribution Panel loads:

  1D10 ckt 04 1D13 125 VDC Distribution Panel C
  1D10 ckt 05 1D14 125 VDC RCIC MCC
  1D10 ckt 06 1D11 125 VDC Distribution Panel A
  1D10 ckt 07 Instrument AC Inverter 1D15 Supply

- 1D11 125 VDC Distribution Panel A loads:

  1D11 ckt 01 Load Center 1B1 switchgear control
  1D11 ckt 02 RWCU F/D Panel 1C82 annunciators
  1D11 ckt 03 Load Center 1B5 switchgear control
  1D11 ckt 04 Load Center 1B7 switchgear control
  1D11 ckt 05 4160V Bus 1A1 switchgear control
  4KV BREAKER 1A101 AUX XFMR TO BUS 1A1
  4KV BREAKER 1A102 STARTUP XFMR TO BUS 1A1
  Reactor Feed Pump 1P-1A Supply Breaker 1A103
  Reactor Recirculation MG Set 1G-201A Supply Breaker 1A104
  Circulating Water Pump 1P-4A Supply Breaker 1A105
  Condensate Pump 1P-8A Supply Breaker 1A106
  FEEDER BREAKER 1A107 1A1 TO LC XFMR 1X11
  FEEDER BREAKER 1A108 1A1 TO LC XFMR 1X71
  FEEDER BREAKER 1A109 1A1 TO LC XFMR 1X51
  FEEDER BREAKER 1A110 1A1 TO SWYD LOAD CENTER
  Reactor pressure Channel B calculation and indication
  Reactor water level Channel B calculation and indication
  1D11 ckt 06 Load Center 1B9 switchgear control
  1D11 ckt 07 MCC breaker 1B3401 control (Normal)
  1D11 ckt 08 Main Generator excitation control
  Generator Exciter Field Breaker control and indication
  Motor Driven DC Regulator Setpoint Adjust (1C08)
  Motor Driven AC Regulator Setpoint Adjust (1C08)
  Exciter Field Flashing
  Regulator Transfer and Lockout Relay
  Exciter Field Bridge Overcurrent Alarm
  Generator Field Bridge Over temperature Alarm
  Exciter Field Current Limit Circuit
  Volts/Hertz Protective Panel
  1D11 ckt 09 Generator H_2 Cooling Panel 1C83
  Associated Generator Trip and Load Runback Relays Annunciators

INFORMATION

- 1D11 125 VDC Distribution Panel A loads (continued):

  - 1D11 ckt 10 Main Transformer 1X1 control power
  - 1D11 ckt 11 1G-31 Diesel Gen. Control Panel 1C117
  - 1D11 ckt 12 1G-31 Diesel Gen. Control Panel 1C117
  - 1D11 ckt 13 Startup Transformer 1X3 control power
  - 1D11 ckt 15 Core Spray Channel A Relay Vertical Board 1C43
    - Core Spray System Loop A Logic
  - 1D11 ckt 17 Radwaste Panel 1C84 annunciators
  - 1D11 ckt 18 1G-31 Diesel Gen. Exciter Panel 1C93
  - 1D11 ckt 19 Standby Gas Treatment System Control Panel 1C24A control
    - PASS System Valves SV-4594A, SV-4595A, and SV-8772A (FC)
    - A SBGTS valves SV-5801A, SV-5815A, SV-5817A, and SV-5825A (CV-5815A, CV-5817A, and CV-5825A (FO) (AV5801 (FC)
    - A SBGTS vent shaft Rad Monitor Aux Relay 95-K134A (PCIS GP 3A input)
    - A SBGTS Fire Deluge SV-5837A (CV-5837A (FC))
    - A SBGTS Preheater control
    - (TORUS) EXTERNAL VACUUM BKR ISOL (CV-4304 (FO))
    - CAMS Loop A Isolation Valve control and indication
    - Offgas Stack Exhaust Fan 1V-EF-18A remote control
    - Panel 1C24A annunciators
    - Panel 1C25A annunciators
  - 1D11 ckt 20 Turbine Building and Control Room HVAC Panel 1C26
    - SFU Fire Deluge SV-7328A (CV-7328A (FC))
    - A DIESEL GENERATOR 1G-31 Room Supply Fan 1V-SF-20 remote control
    - A DIESEL GENERATOR 1G-31 Room Supply Fan 1V-SF-20 dampers DO-7001A and DO-7002A position indication
    - 1V-SFU-30A valves CV-7301A and SV-7318A (AV-7301A and AV-7318A (FO))
    - Miscellaneous Reactor, Turbine, and Control Building isolation dampers
    - 1C23A annunciators
    - 1C26A annunciators and indications

LOSS OF 125 VDC DIV I

INFORMATION

- 1D13 125V DC Distribution Panel C loads:

1D13 ckt 01 Reactor Recirculation Pump MG Set 1G-201A Control Panel 1C112A
and Scoop Tube Power Failure Lock circuitry
1D13 ckt 13 Recirculation Pump MG Set 1G-201A control (Normal and Standby Power)
MG Set A Emergency Lube Oil Pump 1P-204A control and indication
Loss of Division I ATWS/ARI/RPT Trip System (1D1313 only)

1D13 ckt 02 Reactor Core Cooling Benchboard 1C03
RHR Heat Exchanger A Vent Valves MO-2044A and MO-2044B position indication (ZI-2044A and ZI-2044B)
HPCI System Drain Valves SV-2211 and SV-2234 (CV-2211 and CV-2234) control and indication
Position indication for CV-4309 (ZS-4309)

1D13 ckt 03 Reactor Water Cleanup and Recirculation Benchboard 1C04
RCIC Inverter
RCIC Governor Valve HV-2406 Position indication
RCIC System Drain Valves SV-2410, and SV-2435 (CV-2410 and CV-2435) control and indication

1D13 ckt 04 Annunciator Power Panels 1C03, 1C04, 1C05, 1C08B, 1C34, Panel 1C22 Frequency Converter

1D13 ckt 05 CAD Panel 1C35A, CAM Panel 1C09A
SV-4332A, Upper Drywell Spray CAD N2 Primary Containment Isolation
SV-4334A North Torus Spray Header Primary Containment Isol
SV-4332B, and SV-4334B control and indication
1C35A Annunciators
1C09A Annunciators
1C014A EOP Annunciators

1D13 ckt 06 1C40 Annunciators

1D13 ckt 07 1C32 Channel A RHR Relay Vertical Board
RHR Loop A Logic
HPCI Low Water Level initiation signal
HPCI Isolation Logic A
HPCI Rx Hi-Level trip logic (1/2 of logic, HPCI will not trip on Hi-Level)

INFORMATION

- 1D13 125 VDC Distribution Panel C loads:

1D13 ckt 08	Reactor Protection System Channel A Vertical Board 1C15
	Recirculation Pump A Trip circuitry
	Backup Scram Valve SV-1840A (FC)
1D13 ckt 09	Inboard Isolation Valve Relay Panel 1C41
	Inboard MSIVs position indication and DC solenoid valve control
1D13 ckt 10	1C40A Annunciators
1D13 ckt 11	EBO Pump 1P-40 Starter Control (HS-3151) & Indication
	Emergency Bearing Oil Pump 1P-40 control and indication
1D13 ckt 12	Generator and Plant Relay Panel 1C31
	Generator Primary Lockout Relay 286/P
	Startup Transformer - Bus 1A3 Breaker 1A302 Lockout Relay
	Essential Bus 1A3 Load Shedding Circuit
	Non-Essential Auto Transfer and Load Shed
1D13 ckt 14	Auto Blowdown Panel 1C45
	ADS main control power
1D13 ckt 15	4160V Bus 1A3 switchgear control
	4KV BREAKER 1A301 STANDBY TRANSFORMER TO BUS 1A3
	4KV BREAKER 1A302 STARTUP TRANSFORMER TO BUS 1A3
	FEEDER BREAKER 1A303 1A3 TO LC XFMR 1X31
	Core Spray Pump 1P-211A Supply Breaker 1A304
	RHR Pump 1P-229A Supply Breaker 1A305
	RHR Pump 1P-229C Supply Breaker 1A306
	RHR SW Pump 1P-22A Supply Breaker 1A307
	RHR SW Pump 1P-22C Supply Breaker 1A308
	GSW Pump 1P-89A Supply Breaker 1A309
	CRD Pump 1P-209A Supply Breaker 1A310
	A DG TO BUS 1A3 BREAKER 1A311
	FEEDER BREAKER 1A312, 1A3 TO LC XFMR 1X91
	Essential Bus 1A3 Degraded Voltage Detection Circuit
1D13 ckt 16	Load Center 1B3 switchgear control
1D13 ckt 17	RCIC Relay Vertical Board 1C30
	RCIC Turbine Speed Controller
	RCIC Turbine Trip circuitry
	RCIC Initiation and Isolation Relay Logic A
	RCIC Instrumentation
1D13 ckt 19	NSSS Temperature and Leak Detection Panel 1C21
	RCIC Area Steam Leak Detection circuitry Division I
	RCIC Timer Logic Division I
1D13 ckt 20	Remote Shutdown Panels 1C389 and 1C390
	Transfer Switch Position Status Indication

LOSS OF 125 VDC DIV I

INFORMATION

- 1D14 125 VDC RCIC MCC loads:

  1D14 ckt 01 Steam Outboard Isolation Valve MO-2401
  1D14 ckt 02 Steam Supply Valve MO-2404
  1D14 ckt 03 Turbine Stop Valve MO-2405
  1D14 ckt 04 Bypass to Condensate Valve MO-2426
  1D14 ckt 05 Suction from Condensate Storage Tank Valve MO-2500
  1D14 ckt 06 Minimum Flow Bypass Valve MO-2510
  1D14 ckt 07 Normally Open Discharge Valve MO-2511
  1D14 ckt 08 Normally Closed Discharge Valve MO-2512
  1D14 ckt 09 Test Discharge Valve MO-2515
  1D14 ckt 10 Suction at Pool Valve MO-2516
  1D14 ckt 11 Suppression Pool Suction Valve MO-2517
  1D14 ckt 12 Gland Seal Vacuum Pump 1P-227
  1D14 ckt 13 Gland Seal Vacuum Tank Condensate Pump 1P-228

FIGURE F-5
DAEC TELEPHONE SYSTEMS

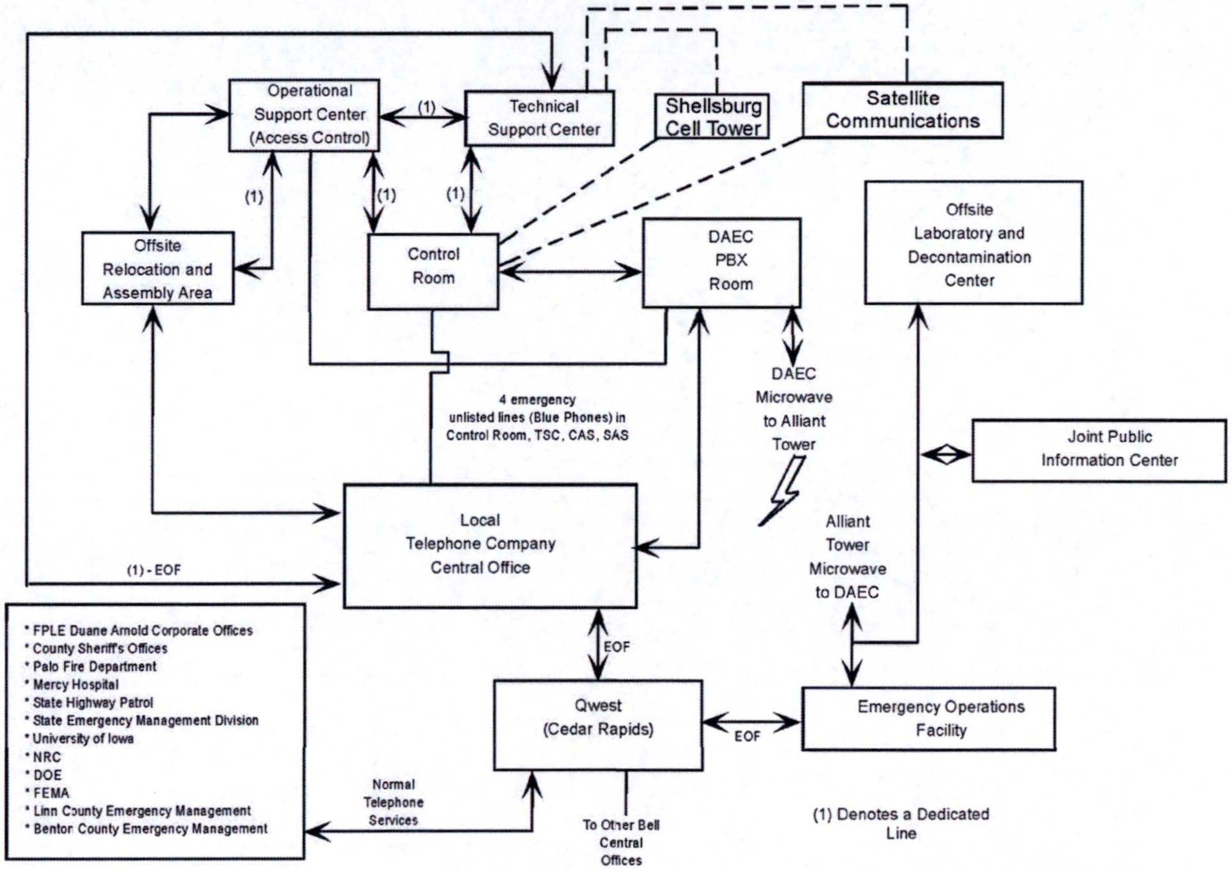


FIGURE F-6
FEDERAL TELEPHONE SYSTEM (FTS-2001)

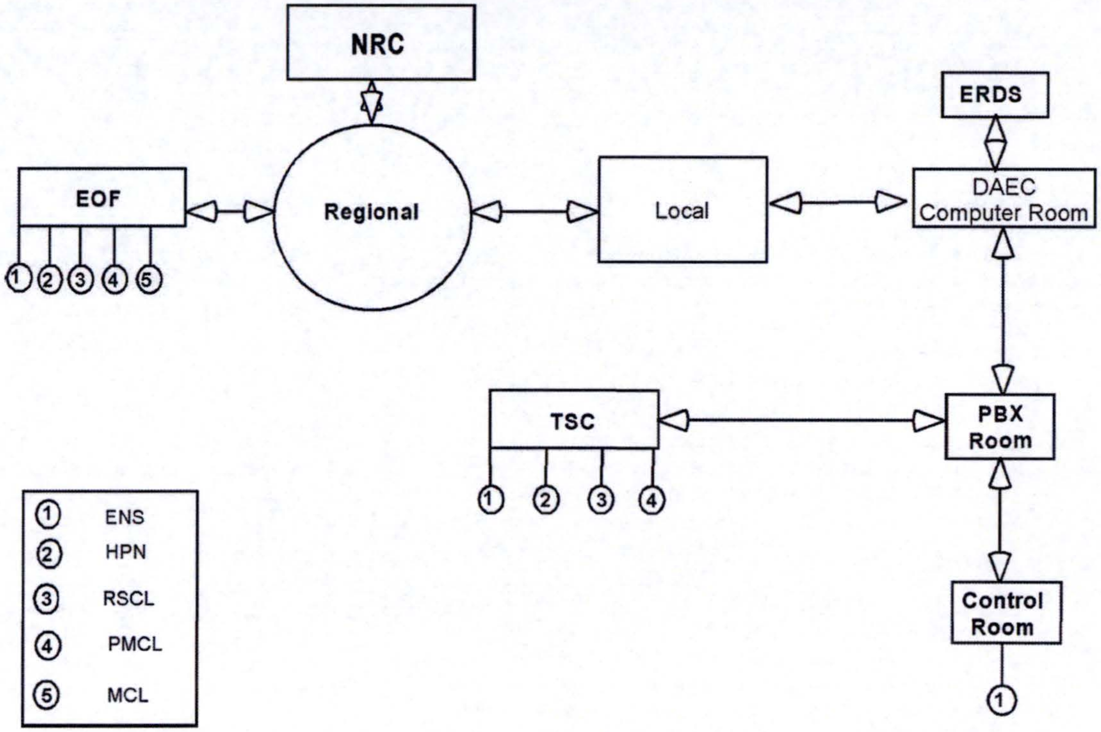
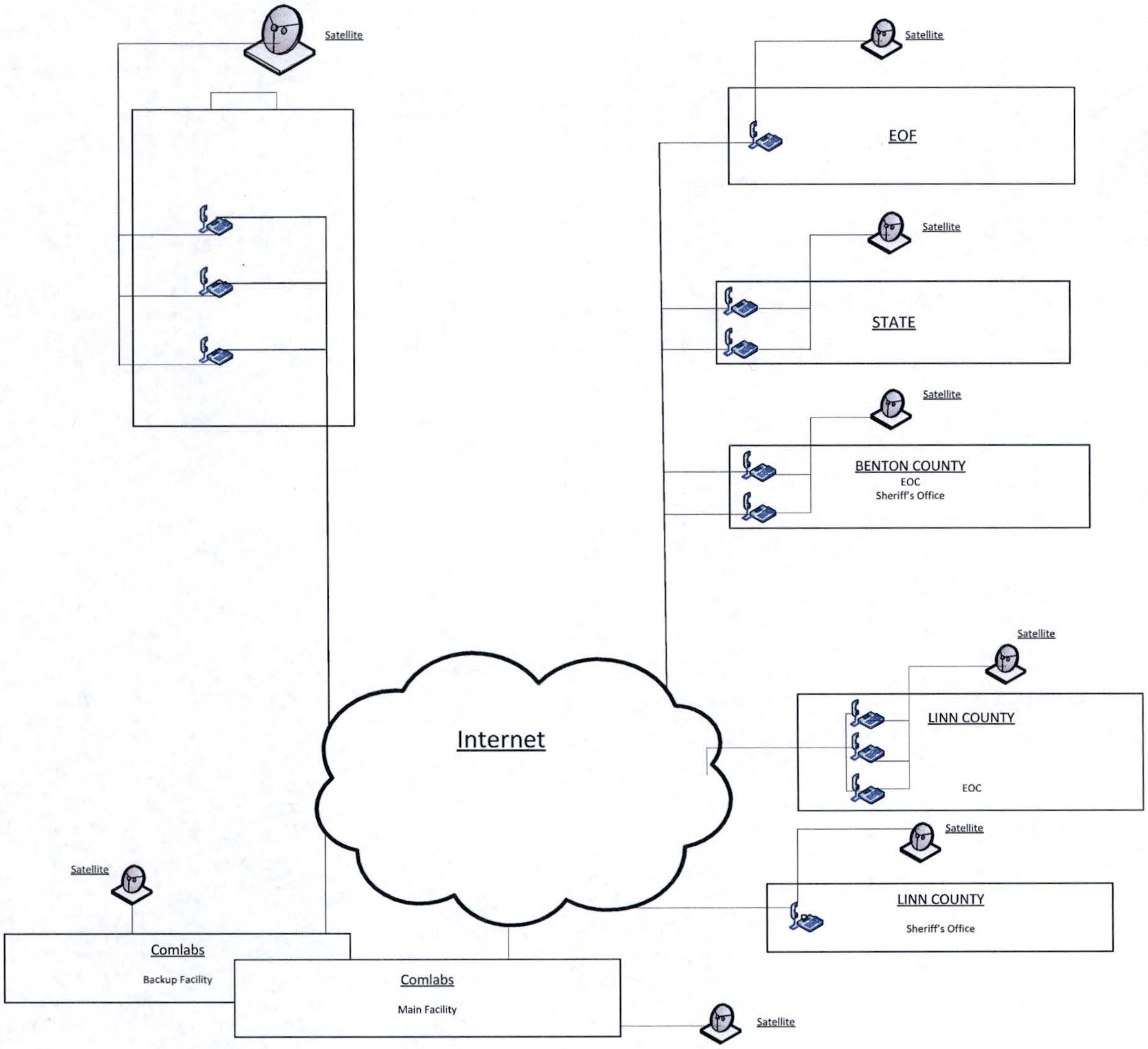


FIGURE F-7
ALL-CALL TELEPHONE SYSTEM



1.0 PURPOSE

(1) This section describes the methods and procedures used by FPLE Duane Arnold to transmit emergency information to the Emergency Response Organization, local and state authorities, and subsequently, from such authorities to the public. Details required in the initial and follow-up message are described, along with a description of the types of news statements that will be used to provide the public with information and protective actions.

2.0 REQUIREMENTS

(1) Methods used to accomplish notification of the Emergency Response Organization include the use of call lists contained in the Emergency Telephone Book, pager and automated telephone callout process.

(2) The Emergency Telephone Book includes phone numbers and pager numbers (where applicable) of emergency response personnel who may be required to respond to an emergency condition. It also includes the 24-hour telephone numbers of local, state, and federal support agencies including the NRC. The NRC would normally be notified using the NRC ENS Telephone (FTS-2001 System) from the Control Room. The state and counties would normally be notified by dedicated microwave telecommunications link.

2.1 INITIAL NOTIFICATION

(1) After declaration of an emergency condition, the Operations Shift Manager/ Supervisor will ensure that the following personnel and agencies are notified:

- Linn and Benton Counties
- State of Iowa
- NRC Operations Center
- Emergency Coordinator
- Emergency Response and Recovery Director
- NRC Resident Inspectors

(2) Verification of Notification

(a) The authenticity of initial notifications provided to Linn and Benton Counties and the State of Iowa do not require verification if the notification is made by the dedicated phone system.

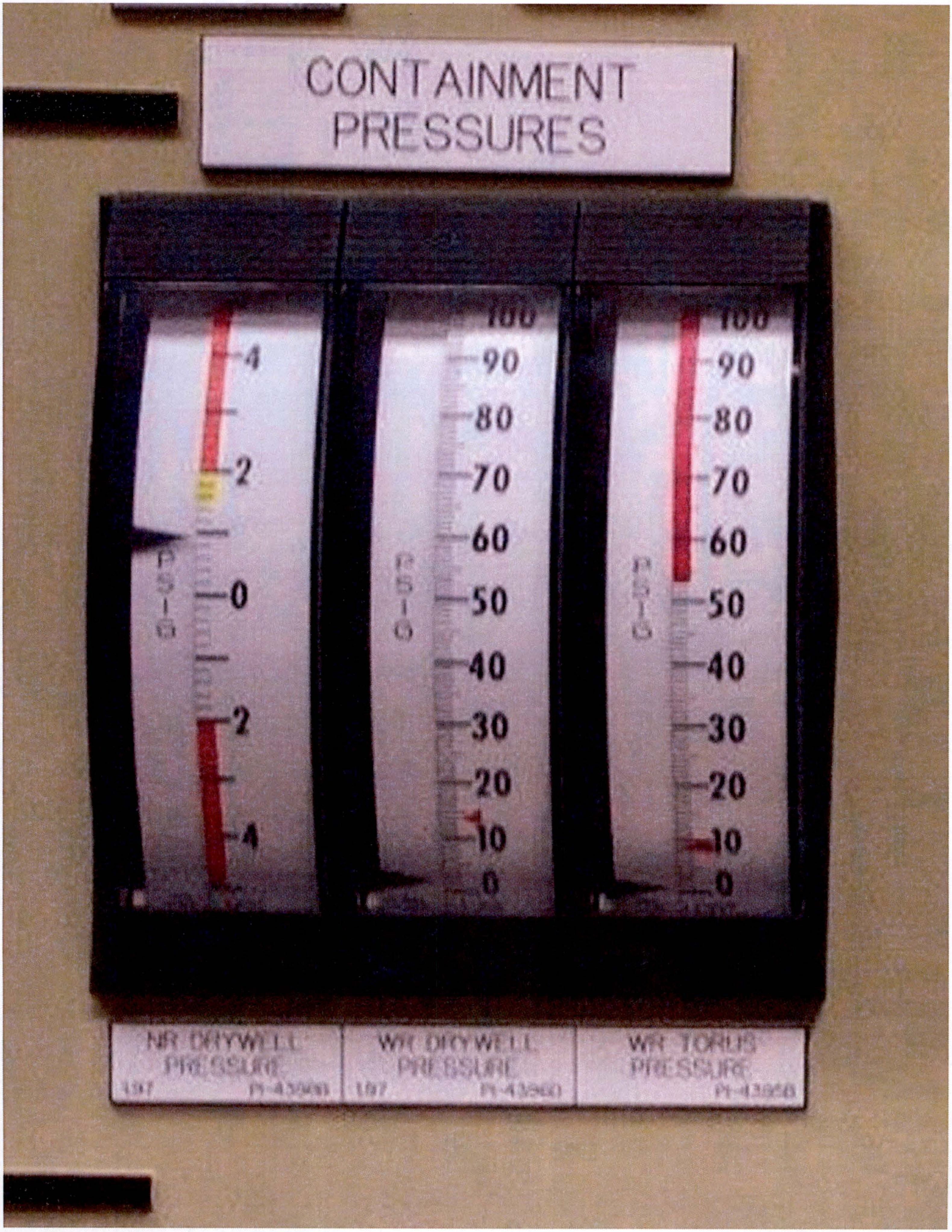
(b) Local and state agencies notified by commercial communication system (telephone or facsimile) may require verification of the identity and authenticity of the caller and the message received.

BREAKPOINTS FOR REACTOR LEVEL CONTROL
Page 1 of 2

RPV Level (inches)	**Item of Interest**	**Significance**
+211	High Level Trip Setpoint, Main Turbine Trip	≠ Loss of high pressure injection (FW, HPCI, RCIC) ≠ Loss of 100% Heat Sink
+170	Low Water Level Scram, PCIS Groups 2, 3, 4 Isolations	≠ RPS defeats needed in ATWS ≠ Containment Isolation, ≠ Shutdown Cooling Valves Close
+119.5	High Pressure Injection, PCIS Group 5 Isolation, ARI	≠ HPCI/RCIC Auto Initiation ≠ RWCU Isolation ≠ ARI Initiation & Recirc Pump ATWS Trip
+87	Two Feet Below Feedwater Sparger	During ATWS if power >5% or unknown, lower level to +87 inches to reduce core inlet subcooling
+64	ECCS Auto Start, PCIS Group 1 Isolation	≠ ADS Timers start ≠ CS/RHR Auto Initiation MSIVs close and result in loss of main condenser
+15	Top of Active Fuel (TAF) (Note 1)	≠ Loss of Adequate Core Cooling (ACC) through core submergence ≠ If no preferred Injection Subsystem is available, maximize injection with Alternate Injection Systems in EOP 1 when level < +15"

Note 1: +15 inches is used for TAF than 0 inches for the following reasons:

- To allow monitoring RPV level on the Wide Range instrumentation - prevents risk of uncovering the core if using Fuel Zone instruments.
- Fuel Zone instruments use the same tap as jet pump instrumentation and any flow through the jet pumps including LPCI flow will cause the Fuel Zone instruments to read high.


CONTAINMENT PRESSURES
PSIG
4
2
0
2
4
PSIG
100
90
80
70
60
50
40
30
20
10
0
PSIG
100
90
80
70
60
50
40
30
20
10
0
NR DRYWELL PRESSURE
197
WR DRYWELL PRESSURE
197
WR TORUS PRESSURE

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

EPFAQ Number:	2016-002
Originator:	David Young
Organization:	NEI
Relevant Guidance:	NEI 99-01, *Methodology for Development of Emergency Action Levels*, Revisions 4 and 5; and NEI 99-01, *Development of Emergency Action Levels for Non-Passive Reactors*, Revision 6. NUMARC/NESP-007, *Methodology for Development of Emergency Action Levels.*
Applicable Section(s):	Initiating Condition (IC) HA2 in NEI 99-01, Revisions 4 and 5, and NUMARC/NESP-007, "FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown" ICs CA6 and SA9 in NEI 99-01, Revision 6: "Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode" Definition of VISIBLE DAMAGE in NEI 99-01, Revisions 4, 5 and 6, and NUMARC/NESP-007
Status:	Complete

NOTE:

Based on NRC staff consideration of industry comments provided by letter dated February 16, 2017 (ADAMS Accession No. ML17079A228), a revision to these ICs was proposed at the public meeting held on April 4, 2017. These changes were attached to the public meeting notice (ADAMS Accession No. ML17089A458). Based on comments provided by the industry during the April 4, 2017 public meeting, the NRC staff revised the proposed revisions to these ICs.

QUESTION OR COMMENT:

A review of industry Operating Experience has identified a need to clarify an aspect of the definition of VISIBLE DAMAGE as it relates to the ICs cited above; adding this clarity is necessary to minimize the potential for an over-classification of an equipment failure. There may be cases where VISIBLE DAMAGE is the result of an equipment failure and limited to the failed component (i.e., the failure did not cause damage to any other component or a structure). The current definition of VISIBLE DAMAGE does not adequately differentiate between damage resulting from, and affecting only, the failed piece of equipment vs. an equipment failure causing damage to another component or a structure (e.g., by a failure-induced fire or explosion). Can the definition of VISIBLE DAMAGE be clarified to help avoid an inappropriate emergency declaration in cases where an equipment failure does not result in damage to another component or a structure (i.e., VISIBLE DAMAGE affects only the failed component)?

A related question is also posed – Consistent with the approach used in other ICs, should a note be added to preclude an emergency declaration if the safety system affected by a hazard was not functional before the event occurred (e.g., tagged out for maintenance)?

PROPOSED SOLUTION:

Yes; the sentence below may be added to the definition of VISIBLE DAMAGE [as defined in NEI 99-01, Revisions 4, 5, and 6].

Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

From a plant safety and change-in-risk perspective, the consequences from the failure of a

piece of equipment, accompanied by a hazard (e.g., a fire or explosion) that does not damage any other equipment or a structure, are essentially the same as the equipment failing with no attendant hazard. Neither event would appear to meet the definition of an Alert because the outcome does not involve an actual or potential substantial degradation of the level of safety of the plant (e.g., there has been no significant reduction in the margin to a loss or potential loss of a fission product barrier). Nuclear power plants are designed with redundant safety system trains that are required to be separated (i.e., installed in separate plant areas or have separation within an individual area).

Absent any collateral damage to another component or a structure, a hazard associated with an equipment failure does not affect the ability to protect public health and safety, and there is no additional response benefit to be gained by declaring an emergency. The normal plant organization has sufficient resources and adequate guidance to respond to an equipment failure – guidance includes operating procedures and Technical Specifications; the fire protection [program], industrial safety and corrective action programs; and work management and maintenance requirements.

Concerning the second question, an emergency declaration would not be appropriate in response to a hazard affecting a piece of equipment or system that was non-functional prior to the event (e.g., tagged out for maintenance). For this reason and consistent with the approach used in other ICs, the following note may be added to IC HA2 (NEI 99-01 R4 and R5), or ICs CA6 and SA9 (NEI 99-01 R6).

Note: If the affected safety system (or component) was already non-functional before the event occurred, then no emergency classification is warranted.

Consistent with the guidance in Regulatory Issue Summary (RIS) 2003-18, Supplement 2, *Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4*, dated January 2003, it is reasonable to conclude that the changes proposed above would be considered as a "deviation."

NRC RESPONSE:

The proposed guidance is intended to ensure that an Alert should be declared only when actual or potential performance issues with SAFETY SYSTEMS have occurred as a result of a hazardous event. The occurrence of a hazardous event will result in a Notification of Unusual Event (NOUE) classification at a minimum. In order to warrant escalation to the Alert classification, the hazardous event should cause indications of degraded performance to one train of a SAFETY SYSTEM with either indications of degraded performance on the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second SAFETY SYSTEM train, such that the operability or reliability of the second train is a concern. In addition, escalation to the Alert classification should not occur if the damage from the hazardous event is limited to a SAFETY SYSTEM that was inoperable, or out of service, prior to the event occurring. As such, the proposed guidance will reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, i.e., does not cause significant concern with shutting down or cooling down the plant.

IC HA2 (NEI 99-01 R4 and R5; NUMARC/NESP-007), or ICs CA6 and SA9 (NEI 99-01 R6), do not directly escalate to a Site Area Emergency or a General Emergency due to a hazardous event. The Fission Product Barrier and/or Abnormal Radiation Levels/Radiological Effluent recognition categories would provide an escalation path to a Site Area Emergency or a General Emergency.

The proposed addition of the following notes, applicable to ICs HA2 (NEI 99-01 R4 and R5; NUMARC/NESP-007), or ICs CA6 and SA9 (NEI 99-01 R6), provide further clarification as to how these Alert emergency classifications are considered. The revisions to these EALs,

including the addition of the notes, are consistent with the current NRC-endorsed Alert classification language.

1. Adding the following note to the applicable EALs, per this EPFAQ, is acceptable as it meets the intent of the EALs, is consistent with other EALs (e.g., EAL HA5 from NEI 99-01, Revision 6; this revision was endorsed by the NRC in a letter dated March 28, 2013, available at ADAMS Accession No. ML12346A463), and ensures that declared emergencies are based upon unplanned events with the potential to pose a radiological risk to the public.

   > *If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.*

2. Adding the following note to help explain the EAL is reasonable to succinctly capture the more detailed information from the Basis section related to when conditions would require the declaration of an Alert.

   > *If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.*

Revising the EALs and the Basis sections to ensure potential escalations from a NOUE to an Alert, due to a hazardous event, is appropriate as the concern with these EALs is: (1) a hazardous event has occurred, (2) one SAFETY SYSTEM train is having performance issues as result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or the VISIBLE DAMAGE is enough to be concerned that the second SAFETY SYSTEM train may have operability or reliability issues.

Revising the definition for VISIBLE DAMAGE is appropriate as this definition is only used for these EALs and the revised EALs are based upon SAFETY SYSTEM trains rather than individual components or structures.

All of the changes discussed above are addressed in the attached markups to NEI 99-01, Revision 6. Licensees that use NESP-007, NEI 99-01 Revision 4, or NEI 99-01 Revision 5 EAL schemes can adopt this language in the relevant format the staff approved for their use.

Consistent with the guidance in Regulatory Issue Summary (RIS) 2003-18, Supplement 2, *Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4*, dated January 2003, a licensee's scheme change based on this EPFAQ should be considered as a "deviation" because a classification based on NRC-endorsed industry guidance in NEI 99-01, Revisions 4, 5 and 6, as well as in NUMARC/NESP-007, could be different from a classification based on this EPFAQ.

RECOMMENDED FUTURE ACTION(S):

☐ INFORMATION ONLY, MAINTAIN EPFAQ

☒ UPDATE GUIDANCE DURING NEXT REVISION

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

CA6

ECL: Alert

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

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

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

AND

b. 1. Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:
    - Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
    - Event damage has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE

such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria 1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance address damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Operators will make a determination of VISIBLE DAMAGE based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

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

Developer Notes:

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

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

ECL Assignment Attributes: 3.1.2.B

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

SA9

ECL: Alert

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode.

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

Example Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

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

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

AND

b. 1. Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:
    - Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
    - Event damage has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE

such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria 1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance address damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Operators will make a determination of VISIBLE DAMAGE based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via ICs FS1 or AS1.

Developer Notes:

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

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

ECL Assignment Attributes: 3.1.2.B

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

VISIBLE DAMAGE: Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM train.

EPFAQ Number:	2018-04
Originator:	David Young
Organization:	NEI
Relevant Guidance:	This question concerns NEI 99-01, *Development of Emergency Action Levels for Non-Passive Reactors*, Revision 6 and EPFAQ 2016-002.
Applicable Section(s):	Initiating Conditions (ICs) CA6 and SA9, and the associated Emergency Action Levels (EALs) and Bases
Date Accepted for Review:	5/31/2018
Status:	Under Review

QUESTION OR COMMENT:

Background

EPFAQ 2016-002 provided guidance intended to reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, i.e., does not cause significant concern with shutting down or cooling down the plant. In responding to the EPFAQ, the staff determined that revising the EALs and the Basis sections of ICs CA6 and SA9 would be appropriate to ensure potential escalations from a NOUE to an Alert, due to a hazardous event, occur when there is: (1) a hazardous event, and (2) one SAFETY SYSTEM train having performance issues as a result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or VISIBLE DAMAGE sufficient to be concerned that the second SAFETY SYSTEM train may have operability or reliability issues. The response to EPFAQ 2016-002 works well for situations involving a safety system with two trains (a typical configuration); however, industry operating experience indicates that additional clarification is needed for three other cases as described in the questions below.

Because this EPFAQ is based on material in EPFAQ 2016-002, the response to this EPFAQ may be considered only by sites that have implemented EPFAQ 2016-002 in a manner approved through an NRC Safety Evaluation Report (SER).

Question

Concerning ICs CA6 and SA9, how should an event leading to indications of degraded performance and/or VISIBLE DAMAGE be classified when:

1. The event affects equipment common to two or more safety systems or safety system trains? For example, a unit with a tank that is the water source for multiple safety injection systems or trains, such as a Refueling Water Storage Tank (RWST).

2. The event affects a safety system that has only one train. For example, a Boiling Water Reactor (BWR) unit with a single-train Reactor Core Isolation Cooling (RCIC) or High-Pressure Coolant Injection (HPCI) system.

3. The event affects two trains of a safety system having more than two trains. For example, a unit that has an Auxiliary/Emergency Feedwater system with three trains.

PROPOSED SOLUTION:

The following answers to the above questions are proposed:

1. An event affecting equipment common to two or more safety systems or safety system trains (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the common equipment) should be classified as an Alert under CA6 or SA9, as appropriate to the plant mode. By affecting the operability or reliability of multiple system trains, the loss of the common equipment effectively meets the two-train impact criteria that underlie the EALs and Bases.

2. An event affecting a single-train safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under CA6 or SA9 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train safety system, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement.

3. An event that affects two trains of a safety system (e.g., one train has indications of degraded performance and the other VISIBLE DAMAGE) that also has one or more additional trains should be classified as an Alert under CA6 or SA9, as appropriate to the plant mode. This approach maintains consistency with the two-train impact criteria that underlie the EALs and Bases, and is warranted because the event was severe enough to affect the operability or reliability of two trains of a safety system despite plant design criteria associated with system and system train separation and protection. Such an event may have caused other plant impacts that are not immediately apparent.

As stated above, this EPFAQ may be considered only by sites that have implemented EPFAQ 2016-002 in a manner approved through an NRC Safety Evaluation Report (SER). With this proviso met, the response to EPFAQ 2018-004 would then provide clarification of expected emergency classifications for cases not explicitly addressed by ICs CA6 and SA9 (from NEI 99- 01, Revision 6), and EPFAQ 2016-002; therefore, implementation of the guidance in this EPFAQ would improve the accuracy and timeliness of a classification following a hazardous event affecting a safety system. Moreover, the answers provided in EPFAQ 2018-004 would result in EAL interpretations that are consistent with the meaning and intent of NRC-approved EAL bases such that the classification of the event would not be different from that approved by the NRC in a site-specific application. For this reason, it is reasonable to conclude that incorporation of the guidance from this EPFAQ into an NRC-approved site-specific scheme reflecting the guidance in EPFAQ 2016-002 would be considered a "difference" in accordance with Regulatory Issue Summary (RIS) 2003-18, Supplement 2, *Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4*, dated January 2003. This "difference" determination is contingent upon incorporating any or all of the three answer statements (as applicable to a facility) verbatim; any change to the scope or intent of the answers would make incorporation into a site-specific scheme a "deviation" per RIS 2003- 018, Supplement 2.

NRC RESPONSE:

RECOMMENDED FUTURE ACTION(S):

☐ INFORMATION ONLY, MAINTAIN EPFAQ

☒ UPDATE GUIDANCE DURING NEXT REVISION

BREAKPOINTS FOR REACTOR LEVEL CONTROL
Page 1 of 2

RPV Level (inches)	Item of Interest	Significance
+211	High Level Trip Setpoint, Main Turbine Trip	• Loss of high pressure injection (FW, HPCI, RCIC) • Loss of 100% Heat Sink
+170	Low Water Level Scram, PCIS Groups 2, 3, 4 Isolations	• RPS defeats needed in ATWS • Containment Isolation, • Shutdown Cooling Valves Close
+119.5	High Pressure Injection, PCIS Group 5 Isolation, ARI	• HPCI/RCIC Auto Initiation • RWCU Isolation • ARI Initiation & Recirc Pump ATWS Trip
+87	Two Feet Below Feedwater Sparger	During ATWS if power >5% or unknown, lower level to +87 inches to reduce core inlet subcooling
+64	ECCS Auto Start, PCIS Group 1 Isolation	• ADS Timers start • CS/RHR Auto Initiation MSIVs close and result in loss of main condenser
+15	Top of Active Fuel (TAF) (Note 1)	• Loss of Adequate Core Cooling (ACC) through core submergence • If no preferred Injection Subsystem is available, maximize injection with Alternate Injection Systems in EOP 1 when level < +15"

Note 1: +15 inches is used for TAF than 0 inches for the following reasons:

- To allow monitoring RPV level on the Wide Range instrumentation - prevents risk of uncovering the core if using Fuel Zone instruments.
- Fuel Zone instruments use the same tap as jet pump instrumentation and any flow through the jet pumps including LPCI flow will cause the Fuel Zone instruments to read high.

SAG-3 HYDROGEN CONTROL

CAUTIONS

3	Operation of HPCI, RCIC, Core Spray, or RHR with suction from the torus and pump flow above the NPSH or vortex limit may damage equipment.
7	H2 and O2 instruments may indicate higher concentrations than actually exist inside the containment following a large break LOCA due to moisture condensation in the sample lines. During the 24 hr period following a large break LOCA, the monitors should not be independently used to support operational decisions but may be used for trending.

START

While in this procedure:

IF	THEN
H_2 or O_2 monitor is unavailable	Notify Chemistry to manually sample the drywell and torus for H_2 and O_2 (PASAP 2.6).
Drywell pressure drops below 2.0 psig	Verify containment sprays isolate.
Torus pressure drops below 2.0 psig	Terminate torus sprays.

H-1

DRYWELL

		Drywell O_2	Drywell O_2	Drywell O_2	Drywell O_2
		< 5%	≥ 5% or unknown		
			Torus H_2		
			None	< 6%	≥ 6% or unknown
Drywell H_2	None	No action	No action	②	③
	< 6%	①	②	②	③
	≥ 6% or unknown	①	③	③	③

7

H-2

TORUS

		Torus O_2	Torus O_2	Torus O_2	Torus O_2
		< 5%	≥ 5% or unknown		
			Drywell H_2		
			None	< 6%	≥ 6% or unknown
Torus H_2	None	No action	No action	⑤	⑥
	< 6%	④	⑤	⑤	⑥
	≥ 6% or unknown	④	⑥	⑥	⑥

7

H-6

Detail D — Normal Release Rate Limits

A determination that the offsite release rate is below normal limits may be made by either:

- Containment Atmosphere Radiation Monitor GASEOUS Channels on-scale and operable.
  - ☛ Monitored locally on RIT-8102A/B at 1C-219A/B or RR-4379A/B at 1C-29 (blue channel).
- Containment sample (PASAP 7.4).

❶

IF........offsite release rate is expected to stay below normal limits (Detail D),
THEN...vent and purge the primary containment:

- ☛ OK to defeat isolations except high radiation (Defeat 9).
- ☛ If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft,
     THEN...vent the drywell through the torus (SEP 301.1).
   - IF........torus water level is at or above 16 ft,
     OR......the torus cannot be vented,
     THEN...vent directly from the drywell (SEP 301.2).
2. IF........the primary containment can be vented,
   THEN...purge the drywell with nitrogen using N_2 purge (SEP 303.2).
3. Stop the vent and purge (if not required by other SAG steps) when:
   - Hydrogen is no longer detected in the drywell,
     OR
   - Offsite release rate reaches normal limits (Detail D).

H-3

❷

IF........offsite release rate is expected to stay below General Emergency Levels (EAL RG1),
OR......RPV water level cannot be maintained above +15 in. (TAF),
THEN...vent and purge the primary containment:

- ☛ OK to defeat all isolations (Defeat 10).
- ☛ If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft,
     THEN...vent the drywell through the torus (SEP 301.1).
   - IF........torus water level is at or above 16 ft,
     OR......the torus cannot be vented,
     THEN...vent directly from the drywell (SEP 301.2).
2. IF........the primary containment can be vented,
   THEN...purge the drywell with nitrogen at max flow using N_2 purge (SEP 303.2).
3. Stop the vent and purge (if not required by other SAG steps) when:
   - Hydrogen is no longer detected in either the drywell or the torus,
     OR
   - Hydrogen is no longer detected in the drywell and drywell oxygen is less than 5%,
     OR
   - RPV water level can be maintained above +15 in. (TAF) and offsite release rate reaches a General Emergency Level (EAL RG1).

H-4

❸

Vent and purge the primary containment:

- ☛ OK to defeat all isolations (Defeat 10).
- ☛ OK to exceed release rate limits.
- ☛ If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft,
     THEN...vent the drywell through the torus (SEP 301.1).
   - IF........torus water level is at or above 16 ft,
     OR......the torus cannot be vented,
     THEN...vent directly from the drywell (SEP 301.2).
2. IF........the primary containment can be vented,
   THEN...purge the drywell with air or nitrogen at max flow. Use whichever method will reduce hydrogen below 6% or oxygen below 5% faster:
   - Air purge (SEP 303.1)
   - N_2 purge (SEP 303.2)
3. IF........permitted by SAG-1,
   THEN...operate drywell sprays (Detail E).

H-5

Detail E — Spray Sources

3

- RHR (OI 149)
- RHR Service Water (AIP 401)
- Fire System (AIP 404)
- Well Water (AIP 403)
- GSW (AIP 403)
- ESW (AIP 402)
- Condensate Service Water (AIP 405)

❹

IF........offsite release rate is expected to stay below normal limits (Detail D),
THEN...vent and purge the primary containment:

- ☛ OK to defeat isolations except high radiation (Defeat 9).
- ☛ If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft,
     THEN...vent directly from the torus (SEP 301.1).
   - IF........the torus cannot be vented,
     AND....torus water level is below 13.5 ft,
     THEN...vent the torus through the drywell (SEP 301.2).
2. IF........the primary containment can be vented,
   THEN...purge the torus with nitrogen using N_2 purge (SEP 303.2).
3. Stop the vent and purge (if not required by other SAG steps) when:
   - Hydrogen is no longer detected in the torus,
     OR
   - Offsite release rate reaches normal limits (Detail D).

H-7

❺

IF........offsite release rate is expected to stay below General Emergency Levels (EAL RG1),
OR......RPV water level cannot be maintained above +15 in. (TAF),
THEN...vent and purge the primary containment:

- ☛ OK to defeat all isolations (Defeat 10).
- ☛ If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft,
     THEN...vent directly from the torus (SEP 301.1).
   - IF........the torus cannot be vented,
     AND....torus water level is below 13.5 ft,
     THEN...vent the torus through the drywell (SEP 301.2).
2. IF........the primary containment can be vented,
   THEN...purge the torus with nitrogen at max flow using N_2 purge (SEP 303.2).
3. Stop the vent and purge (if not required by other SAG steps) when:
   - Hydrogen is no longer detected in either the drywell or the torus,
     OR
   - Hydrogen is no longer detected in the torus and torus oxygen is less than 5%,
     OR
   - RPV water level can be maintained above +15 in. (TAF) and offsite release rate reaches a General Emergency Level (EAL RG1).

H-8

❻

Vent and purge the primary containment:

- ☛ OK to defeat all isolations (Defeat 10).
- ☛ OK to exceed release rate limits.
- ☛ If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1\. IF........permitted by SAG-1,
THEN...operate torus sprays (Detail E).

2\. Vent as follows:
   - IF........torus water level is below 16 ft,
     THEN...vent directly from the torus (SEP 301.1).
   - IF........the torus cannot be vented,
     AND....torus water level is below 13.5 ft,
     THEN...vent the torus through the drywell (SEP 301.2).

2\. IF........the primary containment can be vented,
THEN...purge the torus with air or nitrogen at max flow. Use whichever method will reduce hydrogen below 6% or oxygen below 5% faster:
   - Air purge (SEP 303.1)
   - N_2 purge (SEP 303.2)

H-9

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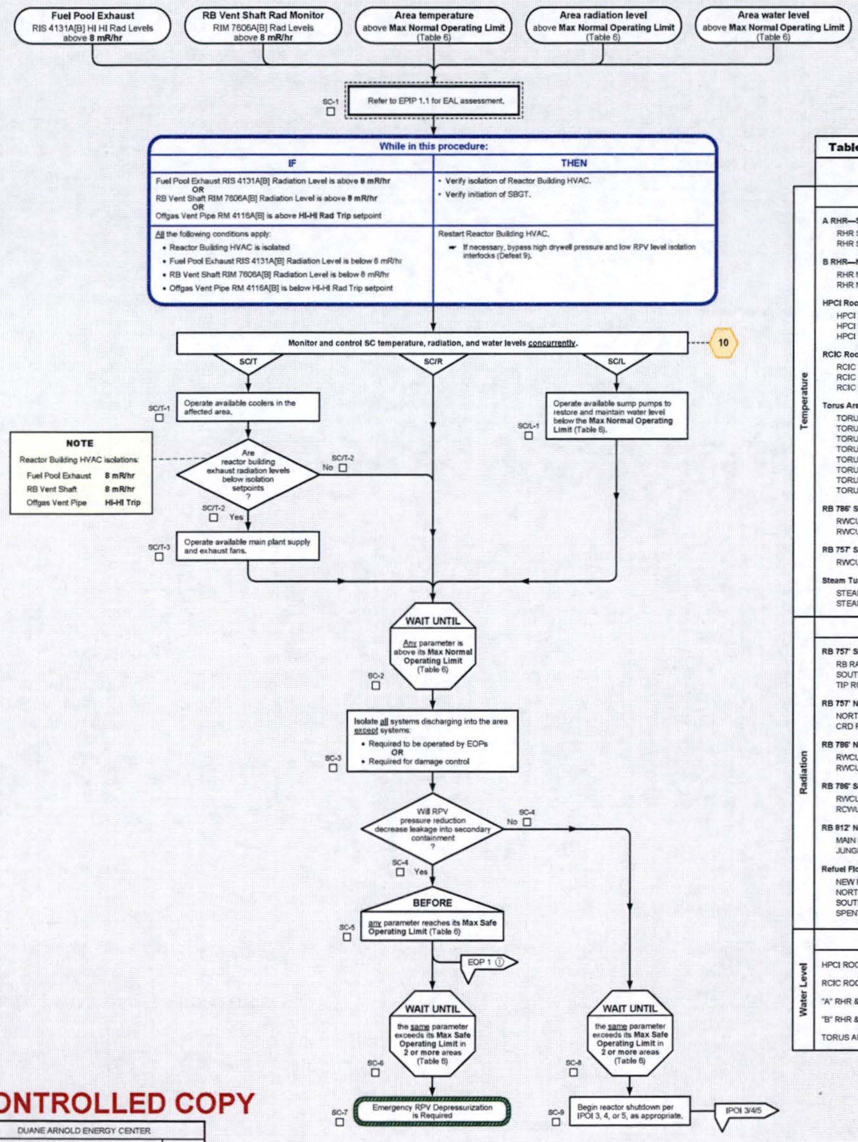
SAG-3 HYDROGEN CONTROL	REV. 6

EOP 3
SECONDARY CONTAINMENT CONTROL



Fuel Pool Exhaust
RIS 4131A[B] HI HI Rad Levels above **8 mR/hr**

RB Vent Shaft Rad Monitor
RIM 7606A[B] Rad Levels above **8 mR/hr**

Area temperature
above **Max Normal Operating Limit** (Table 6)

Area radiation level
above **Max Normal Operating Limit** (Table 6)

Area water level
above **Max Normal Operating Limit** (Table 6)

SC-1 ☐ Refer to EPIP 1.1 for EAL assessment.

While in this procedure:

IF	THEN
Fuel Pool Exhaust RIS 4131A[B] Radiation Level is above **8 mR/hr** **OR** RB Vent Shaft RIM 7606A[B] Radiation Level is above **8 mR/hr** **OR** Offgas Vent Pipe RM 4116A[B] is above **HI-HI Rad Trip** setpoint	• Verify isolation of Reactor Building HVAC. • Verify initiation of SBGT.
All the following conditions apply: • Reactor Building HVAC is isolated • Fuel Pool Exhaust RIS 4131A[B] Radiation Level is below 8 mR/hr • RB Vent Shaft RIM 7606A[B] Radiation Level is below 8 mR/hr • Offgas Vent Pipe RM 4116A[B] is below HI-HI Rad Trip setpoint	Restart Reactor Building HVAC. ➡ If necessary, bypass high drywell pressure and low RPV level isolation interlocks (Defeat 9).

Monitor and control SC temperature, radiation, and water levels concurrently. — 10

SC/T

SC/T-1 ☐ Operate available coolers in the affected area.

NOTE

Reactor Building HVAC isolations:

Fuel Pool Exhaust	**8 mR/hr**
RB Vent Shaft	**8 mR/hr**
Offgas Vent Pipe	**HI-HI Trip**

Are reactor building exhaust radiation levels below isolation setpoints?

SC/T-2 No ☐

SC/T-2 ☐ Yes

SC/T-3 ☐ Operate available main plant supply and exhaust fans.

SC/R

SC/L

SC/L-1 ☐ Operate available sump pumps to restore and maintain water level below the **Max Normal Operating Limit** (Table 6).

WAIT UNTIL
Any parameter is above its **Max Normal Operating Limit** (Table 6)
SC-2 ☐

SC-3 ☐ Isolate all systems discharging into the area except systems:
- Required to be operated by EOPs
  OR
- Required for damage control

Will RPV pressure reduction decrease leakage into secondary containment?

SC-4 No ☐

SC-4 ☐ Yes

BEFORE
any parameter reaches its **Max Safe Operating Limit** (Table 6)
SC-5 ☐

EOP 1 ①

WAIT UNTIL
the same parameter exceeds its **Max Safe Operating Limit** in **2 or more areas** (Table 6)
SC-6 ☐

SC-7 ☐ Emergency RPV Depressurization is Required

WAIT UNTIL
the same parameter exceeds its **Max Safe Operating Limit** in **2 or more areas** (Table 6)
SC-8 ☐

SC-9 ☐ Begin reactor shutdown per IPOI 3, 4, or 5, as appropriate.

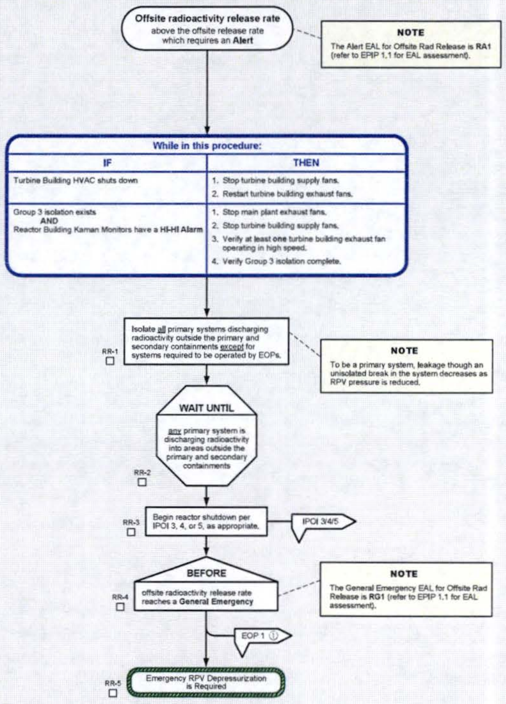
IPOI 3/4/5

CAUTIONS

10 High spent fuel pool temperature or low spent fuel pool water level may affect conditions within secondary containment.

Table 6	**Secondary Containment Limits**				
	Parameter and Areas		**Max Normal Operating Limit**	**Max Safe Operating Limit**	**Value/Trend**
	Area/Location	Indicator	°F	°F	
Temperature	**A RHR—SE Corner Room Area**				
	RHR SE CORNER ROOM AMBIENT	TR/TDR 2000A Ch 1	130	140	
	RHR SE CORNER ROOM DIFFERENTIAL	TR/TDR 2000A Ch 2	50	N/A	
	B RHR—NW Corner Room Area				
	RHR NW CORNER ROOM AMBIENT	TR/TDR 2000B Ch 1	130	140	
	RHR NW CORNER ROOM DIFFERENTIAL	TR/TDR 2000B Ch 2	50	N/A	
	HPCI Room Area				
	HPCI EMER COOLER AMBIENT	TR/TDR 2225A[B] Ch 1	175	310	
	HPCI ROOM AMBIENT	TR/TDR 2225A Ch 2	175	310	
	HPCI ROOM DIFFERENTIAL	TR/TDR 2225A[B] Ch 4[3]	50	N/A	
	RCIC Room Area				
	RCIC EMER COOLER AMBIENT	TR/TDR 2425A[B] Ch 1	175	300	
	RCIC ROOM AMBIENT	TR/TDR 2425A Ch 2	175	300	
	RCIC ROOM DIFFERENTIAL	TR/TDR 2425A[B] Ch 4	50	N/A	
	Torus Area				
	TORUS CATWALK NORTH AMBIENT	TR/TDR 2425A Ch 3	150	165	
	TORUS CATWALK WEST AMBIENT	TR/TDR 2425B Ch 2	150	165	
	TORUS CATWALK SOUTH AMBIENT	TR/TDR 2225A Ch 3	150	165	
	TORUS CATWALK EAST AMBIENT	TR/TDR 2225B Ch 2	150	165	
	TORUS CATWALK EAST DIFF	TR/TDR 2425A Ch 5	50	N/A	
	TORUS CATWALK WEST DIFF	TR/TDR 2425B Ch 5	50	N/A	
	TORUS CATWALK SOUTHWEST DIFF	TR/TDR 2225A Ch 5	50	N/A	
	TORUS CATWALK SOUTH DIFF	TR/TDR 2225B Ch 4	50	N/A	
	RB 786' South Area				
	RWCU PUMP ROOM AMBIENT	TR/TDR 2700A[B] Ch 1	130	212	
	RWCU HX ROOM AMBIENT	TR/TDR 2700A[B] Ch 2,3	130	212	
	RB 757' South Area				
	RWCU ABOVE TIP ROOM AMBIENT	TR/TDR 2700A[B] Ch 4,5	111.5	150	
	Steam Tunnel Area				
	STEAM TUNNEL AMBIENT	TR/TDR 2425B Ch 3	160	300	
	STEAM TUNNEL DIFFERENTIAL	TR/TDR 2225B Ch 5	70	N/A	
Radiation	Area/Location	Indicator	mR/hr	mR/hr	
	RB 757' South Area				
	RB RAILROAD ACCESS AREA	RI 9167	10	100	
	SOUTH CRD MODULE AREA	RI 9169	10	100	
	TIP ROOM	RI 9176	60	600	
	RB 757' North Area				
	NORTH CRD MODULE	RI 9168	10	100	
	CRD REPAIR ROOM	RI 9170	15	150	
	RB 786' North Area				
	RWCU SPENT RESIN ROOM	RI 9173	100	10^3	
	RWCU PHASE SEP TANK ROOM	RI 9177	20	200	
	RB 786' South Area				
	RWCU PUMP ROOM	RI 9156	10^3	10^4	
	RCWU HX ROOM	RI 9157	10^3	10^4	
	RB 812' North Area				
	MAIN PLANT EXHAUST FAN ROOM	RI 9171	60	600	
	JUNGLE ROOM	RI 9155	60	600	
	Refuel Floor Area				
	NEW FUEL VAULT AREA	RI 9153	10	100	
	NORTH REFUEL FLOOR	RI 9163	10	100	
	SOUTH REFUEL FLOOR	RI 9164	10	100	
	SPENT FUEL POOL AREA	RI 9178	100	10^3	
Water Level	Area/Location	Indicator	inches	inches	
	HPCI ROOM	LI 3768	2	6	
	RCIC ROOM	LI 3769	3	6	
	"A" RHR & CS SECR	LI 3770	2	10	
	"B" RHR & CS NWCR	LI 3771	2	10	
	TORUS AREA	LI 3772	2	12	

EOP 4
RADIOACTIVITY RELEASE CONTROL



Offsite radioactivity release rate
above the offsite release rate which requires an **Alert**

NOTE

The Alert EAL for Offsite Rad Release is **RA1** (refer to EPIP 1.1 for EAL assessment).

While in this procedure:

IF	THEN
Turbine Building HVAC shuts down	1. Stop turbine building supply fans. 2. Restart turbine building exhaust fans.
Group 3 isolation exists **AND** Reactor Building Kaman Monitors have a **HI-HI Alarm**	1. Stop main plant exhaust fans. 2. Stop turbine building supply fans. 3. Verify **at least one** turbine building exhaust fan operating in high speed. 4. Verify Group 3 isolation complete.

RR-1 ☐ Isolate all primary systems discharging radioactivity outside the primary and secondary containments except for systems required to be operated by EOPs.

NOTE

To be a primary system, leakage though an unisolated break in the system decreases as RPV pressure is reduced.

WAIT UNTIL
any primary system is discharging radioactivity into areas outside the primary and secondary containments
RR-2 ☐

RR-3 ☐ Begin reactor shutdown per IPOI 3, 4, or 5, as appropriate.

IPOI 3/4/5

BEFORE
offsite radioactivity release rate reaches a **General Emergency**
RR-4 ☐

NOTE

The General Emergency EAL for Offsite Rad Release is **RG1** (refer to EPIP 1.1 for EAL assessment).

EOP 1 ①

RR-5 ☐ Emergency RPV Depressurization is Required

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EOP 3 – SECONDARY CONTAINMENT CONTROL EOP 4 – RADIOACTIVITY RELEASE CONTROL	REV. 21

1.2.7 HSM Dose Rates with a Loaded 24P, 52B or 61BT DSC

Limit/Specification:

Dose rates at the following locations shall be limited to levels which are less than or equal to:

a. 400 mrem/hr at 3 feet from the HSM surface.

b. Outside of HSM door on center line of DSC 100 mrem/hr.

c. End shield wall exterior 20 mrem/hr.

Applicability: This specification is applicable to all HSMs which contain a loaded 24P, 52B or 61BT DSC.

Objective: The dose rate is limited to this value to ensure that the cask (DSC) has not been inadvertently loaded with fuel not meeting the specifications in Section 1.2.1 and to maintain dose rates as-low-as-is-reasonably achievable (ALARA) at locations on the HSMs where surveillance is performed, and to reduce off-site exposures during storage.

Action:

a. If specified dose rates are exceeded, the following actions should be taken:

61BT DSC Dose Rate
Thresholds = 2 X TS limits

Therefore:

3 feet from HSM Surface
= 800 mrem/hr

Outside HSM Door –
Centerline of DSC
= 200 mrem/hr

End Shield Wall Exterior
= 40 mrem/hr

1. Ensure that the DSC is properly positioned on the support rails.
2. Ensure proper installation of the HSM door.
3. Ensure that the required module spacing is maintained.
4. Confirm that the spent fuel assemblies contained in the DSC conform to the specifications of Section 1.2.1.
5. Install temporary or permanent shielding to mitigate the dose to acceptable levels in accordance with 10 CFR Part 20, 10 CFR 72.104(a), and ALARA.

b. Submit a letter report to the NRC within 30 days summarizing the action taken and the results of the surveillance, investigation and findings. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to the administrator of the appropriate NRC regional office.

Surveillance: The HSM and ISFSI shall be checked to verify that this specification has been met after the DSC is placed into storage and the HSM door is closed.

Basis: The basis for this limit is the shielding analysis presented in Section 7.0, Appendix J, and Appendix K of the FSAR. The specified dose rates provide as-low-as-is-reasonably-achievable on-site and off-site doses in accordance with 10 CFR Part 20 and 10 CFR 72.104(a).

Development of EAL Threshold values from NEE-323-CALC-001

Calculated values were added to the typical background readings of these monitors, and then rounded to aid in evaluator use of the EALs.

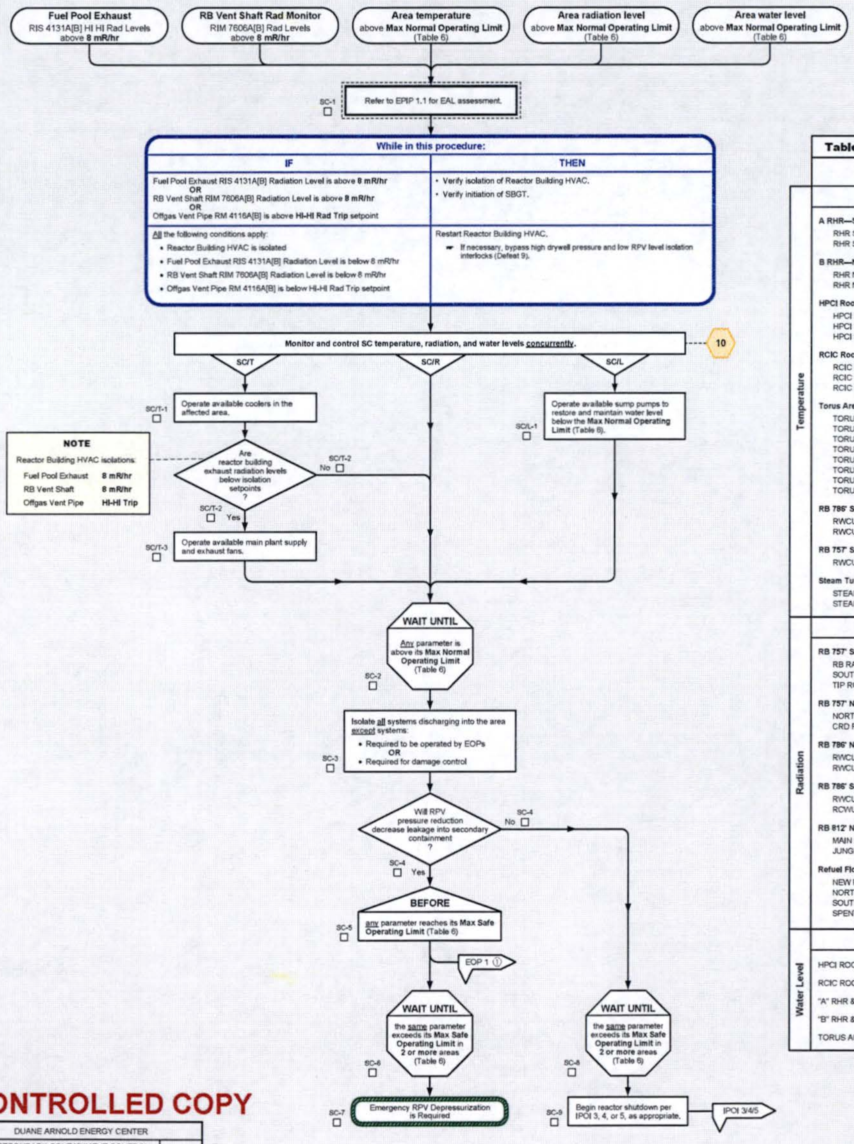
Resulting values used in the DAEC Fission Product Barrier chart are shown below:

- **Fuel Clad Barrier**:
  - Fuel Clad Barrier LOSS 4.A = Drywell Monitor (9184A/B) reading greater than 2000 R/hr.
  - Fuel Clad Barrier LOSS 4.B = Torus Monitor (9185A/B) reading greater than 200 R/hr.
- **RCS Barrier**:
  - • RCS Barrier LOSS 4.A = Drywell Monitor (9184A/B) reading greater than 5 R/hr after reactor shutdown. (minimum serviceable threshold value accounting for scale of monitor)
  - • Calculated Torus Monitor (9185A/B) response is below scale of monitor and not used.
- **CTMT Barrier**:
  - • CTMT Barrier LOSS 4.A = Drywell Monitor (9184A/B) reading greater than 5000 R/hr.
  - • CTMT Barrier LOSS 4.A = Torus Monitor (9185A/B) reading greater than 500 R/hr.

BREAKPOINTS FOR PRIMARY CONTAINMENT PRESSURE CONTROL

Pressure (psig)	Item of Interest	Significance
53 (Torus)	Primary Containment Pressure Limit (PCPL)	When PCPL is reached, containment venting is required.
~21.4 (Torus)	Pressure Suppression	Pressure Suppression Pressure exceeded for normal torus level
>11 (Torus) (11.15)	Drywell Sprays	Drywell sprays may be initiated if drywell parameters are within the Drywell Spray Initiation Limit and torus level is less than 13.5 feet
11.4 (Drywell)	Drywell Spray Initiation Limit (DWSIL) Break Point	Above 11.4 psig drywell pressure, drywell spray initiation is unrestricted by the DWSIL.
<11 (Torus) (11.15)	Torus Spray Initiation Pressure	Start torus sprays prior to 11 psig, if possible. If pressure is exceeded before torus sprays are initiated - initiate them anyway
2 (Drywell)	Drywell High Pressure Scram Setpoint	ECCS Initiation, Isolations and RPS defeats may be needed, EOP 1 and EOP 2 entry
1 (Drywell)	Drywell N2 Makeup Isolation	Drywell N2 makeup supply isolates if drywell pressure exceeds 1 psig

EOP 3
SECONDARY CONTAINMENT CONTROL



Fuel Pool Exhaust
RIS 4131A[B] HI HI Rad Levels
above **8 mR/hr**

RB Vent Shaft Rad Monitor
RIM 7606A[B] Rad Levels
above **8 mR/hr**

Area temperature
above **Max Normal Operating Limit**
(Table 6)

Area radiation level
above **Max Normal Operating Limit**
(Table 6)

Area water level
above **Max Normal Operating Limit**
(Table 6)

SC-1 ☐ Refer to EPIP 1.1 for EAL assessment.

While in this procedure:

IF	THEN
Fuel Pool Exhaust RIS 4131A[B] Radiation Level is above **8 mR/hr** **OR** RB Vent Shaft RIM 7606A[B] Radiation Level is above **8 mR/hr** **OR** Offgas Vent Pipe RM 4116A[B] is above **HI-HI Rad Trip** setpoint	• Verify isolation of Reactor Building HVAC. • Verify initiation of SBGT.
All the following conditions apply: • Reactor Building HVAC is isolated • Fuel Pool Exhaust RIS 4131A[B] Radiation Level is below 8 mR/hr • RB Vent Shaft RIM 7606A[B] Radiation Level is below 8 mR/hr • Offgas Vent Pipe RM 4116A[B] is below HI-HI Rad Trip setpoint	Restart Reactor Building HVAC. ➨ If necessary, bypass high drywell pressure and low RPV level isolation interlocks (Defeat 9).

Monitor and control SC temperature, radiation, and water levels concurrently. ⑩

SC/T

SC/T-1 ☐ Operate available coolers in the affected area.

NOTE

Reactor Building HVAC isolations:

Fuel Pool Exhaust	**8 mR/hr**
RB Vent Shaft	**8 mR/hr**
Offgas Vent Pipe	**HI-HI Trip**

Are reactor building exhaust radiation levels below isolation setpoints ?

SC/T-2 No ☐

SC/T-2 ☐ Yes

SC/T-3 ☐ Operate available main plant supply and exhaust fans.

SC/R

SC/L

SC/L-1 ☐ Operate available sump pumps to restore and maintain water level below the **Max Normal Operating Limit** (Table 6).

WAIT UNTIL
Any parameter is above its **Max Normal Operating Limit** (Table 6)
SC-2 ☐

SC-3 ☐ Isolate all systems discharging into the area except systems:
- Required to be operated by EOPs
  OR
- Required for damage control

Will RPV pressure reduction decrease leakage into secondary containment ?

SC-4 No ☐

SC-4 ☐ Yes

BEFORE
SC-5 ☐ any parameter reaches its **Max Safe Operating Limit** (Table 6)

EOP 1 ①

WAIT UNTIL
the same parameter exceeds its **Max Safe Operating Limit** in **2 or more areas** (Table 6)
SC-6 ☐

SC-7 ☐ Emergency RPV Depressurization is Required

WAIT UNTIL
the same parameter exceeds its **Max Safe Operating Limit** in **2 or more areas** (Table 6)
SC-8 ☐

SC-9 ☐ Begin reactor shutdown per IPOI 3, 4, or 5, as appropriate.

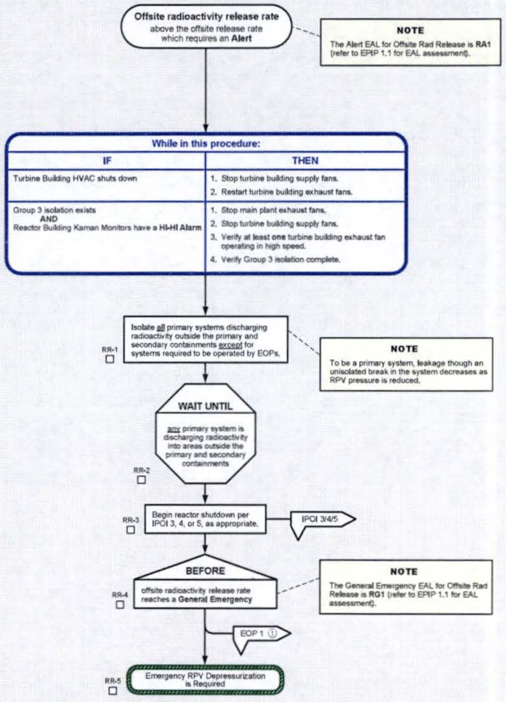
IPOI 3/4/5

CAUTIONS

⑩ High spent fuel pool temperature or low spent fuel pool water level may affect conditions within secondary containment.

Table 6	**Secondary Containment Limits**				
	Parameter and Areas		**Max Normal Operating Limit**	**Max Safe Operating Limit**	**Value/Trend**
Temperature	Area/Location	Indicator	°F	°F	
	A RHR—SE Corner Room Area				
	RHR SE CORNER ROOM AMBIENT	TR/TDR 2000A Ch 1	130	140	
	RHR SE CORNER ROOM DIFFERENTIAL	TR/TDR 2000A Ch 2	50	N/A	
	B RHR—NW Corner Room Area				
	RHR NW CORNER ROOM AMBIENT	TR/TDR 2000B Ch 1	130	140	
	RHR NW CORNER ROOM DIFFERENTIAL	TR/TDR 2000B Ch 2	50	N/A	
	HPCI Room Area				
	HPCI EMER COOLER AMBIENT	TR/TDR 2225A[B] Ch 1	175	310	
	HPCI ROOM AMBIENT	TR/TDR 2225A Ch 2	175	310	
	HPCI ROOM DIFFERENTIAL	TR/TDR 2225A[B] Ch 4[3]	50	N/A	
	RCIC Room Area				
	RCIC EMER COOLER AMBIENT	TR/TDR 2425A[B] Ch 1	175	300	
	RCIC ROOM AMBIENT	TR/TDR 2425A Ch 2	175	300	
	RCIC ROOM DIFFERENTIAL	TR/TDR 2425A[B] Ch 4	50	N/A	
	Torus Area				
	TORUS CATWALK NORTH AMBIENT	TR/TDR 2425A Ch 3	150	165	
	TORUS CATWALK WEST AMBIENT	TR/TDR 2425B Ch 2	150	165	
	TORUS CATWALK SOUTH AMBIENT	TR/TDR 2225A Ch 3	150	165	
	TORUS CATWALK EAST AMBIENT	TR/TDR 2225B Ch 2	150	165	
	TORUS CATWALK EAST DIFF	TR/TDR 2425A Ch 5	50	N/A	
	TORUS CATWALK WEST DIFF	TR/TDR 2425B Ch 5	50	N/A	
	TORUS CATWALK SOUTHWEST DIFF	TR/TDR 2225A Ch 5	50	N/A	
	TORUS CATWALK SOUTH DIFF	TR/TDR 2225B Ch 4	50	N/A	
	RB 786' South Area				
	RWCU PUMP ROOM AMBIENT	TR/TDR 2700A[B] Ch 1	130	212	
	RWCU HX ROOM AMBIENT	TR/TDR 2700A[B] Ch 2,3	130	212	
	RB 757' South Area				
	RWCU ABOVE TIP ROOM AMBIENT	TR/TDR 2700A[B] Ch 4,5	111.5	150	
	Steam Tunnel Area				
	STEAM TUNNEL AMBIENT	TR/TDR 2425B Ch 3	160	300	
	STEAM TUNNEL DIFFERENTIAL	TR/TDR 2225B Ch 5	70	N/A	
Radiation	Area/Location	Indicator	mR/hr	mR/hr	
	RB 757' South Area				
	RB RAILROAD ACCESS AREA	RI 9167	10	100	
	SOUTH CRD MODULE AREA	RI 9169	10	100	
	TIP ROOM	RI 9176	60	600	
	RB 757' North Area				
	NORTH CRD MODULE	RI 9168	10	100	
	CRD REPAIR ROOM	RI 9170	15	150	
	RB 786' North Area				
	RWCU SPENT RESIN ROOM	RI 9173	100	10^3	
	RWCU PHASE SEP TANK ROOM	RI 9177	20	200	
	RB 786' South Area				
	RWCU PUMP ROOM	RI 9156	10^3	10^4	
	RCWU HX ROOM	RI 9157	10^3	10^4	
	RB 812' North Area				
	MAIN PLANT EXHAUST FAN ROOM	RI 9171	60	600	
	JUNGLE ROOM	RI 9155	60	600	
	Refuel Floor Area				
	NEW FUEL VAULT AREA	RI 9153	10	100	
	NORTH REFUEL FLOOR	RI 9163	10	100	
	SOUTH REFUEL FLOOR	RI 9164	10	100	
	SPENT FUEL POOL AREA	RI 9178	100	10^3	
Water Level	Area/Location	Indicator	inches	inches	
	HPCI ROOM	LI 3768	2	6	
	RCIC ROOM	LI 3769	3	6	
	"A" RHR & CS SECR	LI 3770	2	10	
	"B" RHR & CS NWCR	LI 3771	2	10	
	TORUS AREA	LI 3772	2	12	

EOP 4
RADIOACTIVITY RELEASE CONTROL



Offsite radioactivity release rate
above the offsite release rate
which requires an **Alert**

NOTE

The Alert EAL for Offsite Rad Release is **RA1** (refer to EPIP 1.1 for EAL assessment).

While in this procedure:

IF	THEN
Turbine Building HVAC shuts down	1. Stop turbine building supply fans. 2. Restart turbine building exhaust fans.
Group 3 isolation exists **AND** Reactor Building Kaman Monitors have a **HI-HI Alarm**	1. Stop main plant exhaust fans. 2. Stop turbine building supply fans. 3. Verify **at least one** turbine building exhaust fan operating in high speed. 4. Verify Group 3 isolation complete.

RR-1 ☐ Isolate all primary systems discharging radioactivity outside the primary and secondary containments except for systems required to be operated by EOPs.

NOTE

To be a primary system, leakage though an unisolated break in the system decreases as RPV pressure is reduced.

WAIT UNTIL
any primary system is discharging radioactivity into areas outside the primary and secondary containments
RR-2 ☐

RR-3 ☐ Begin reactor shutdown per IPOI 3, 4, or 5, as appropriate.

IPOI 3/4/5

BEFORE
RR-4 ☐ offsite radioactivity release rate reaches a **General Emergency**

NOTE

The General Emergency EAL for Offsite Rad Release is **RG1** (refer to EPIP 1.1 for EAL assessment).

EOP 1 ①

RR-5 ☐ Emergency RPV Depressurization is Required

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DUANE ARNOLD ENERGY CENTER	
EOP 3 – SECONDARY CONTAINMENT CONTROL EOP 4 – RADIOACTIVITY RELEASE CONTROL	REV. 21

BREAKPOINTS FOR PRIMARY CONTAINMENT PRESSURE CONTROL

Pressure (psig)	Item of Interest	Significance
53 (Torus)	Primary Containment Pressure Limit (PCPL)	When PCPL is reached, containment venting is required.
~21.4 (Torus)	Pressure Suppression	Pressure Suppression Pressure exceeded for normal torus level
>11 (Torus) (11.15)	Drywell Sprays	Drywell sprays may be initiated if drywell parameters are within the Drywell Spray Initiation Limit and torus level is less than 13.5 feet
11.4 (Drywell)	Drywell Spray Initiation Limit (DWSIL) Break Point	Above 11.4 psig drywell pressure, drywell spray initiation is unrestricted by the DWSIL.
<11 (Torus) (11.15)	Torus Spray Initiation Pressure	Start torus sprays prior to 11 psig, if possible. If pressure is exceeded before torus sprays are initiated - initiate them anyway
2 (Drywell)	Drywell High Pressure Scram Setpoint	ECCS Initiation, Isolations and RPS defeats may be needed, EOP 1 and EOP 2 entry
1 (Drywell)	Drywell N2 Makeup Isolation	Drywell N2 makeup supply isolates if drywell pressure exceeds 1 psig

EOP 2 – PRIMARY CONTAINMENT CONTROL

Torus water level below **10.1 ft**

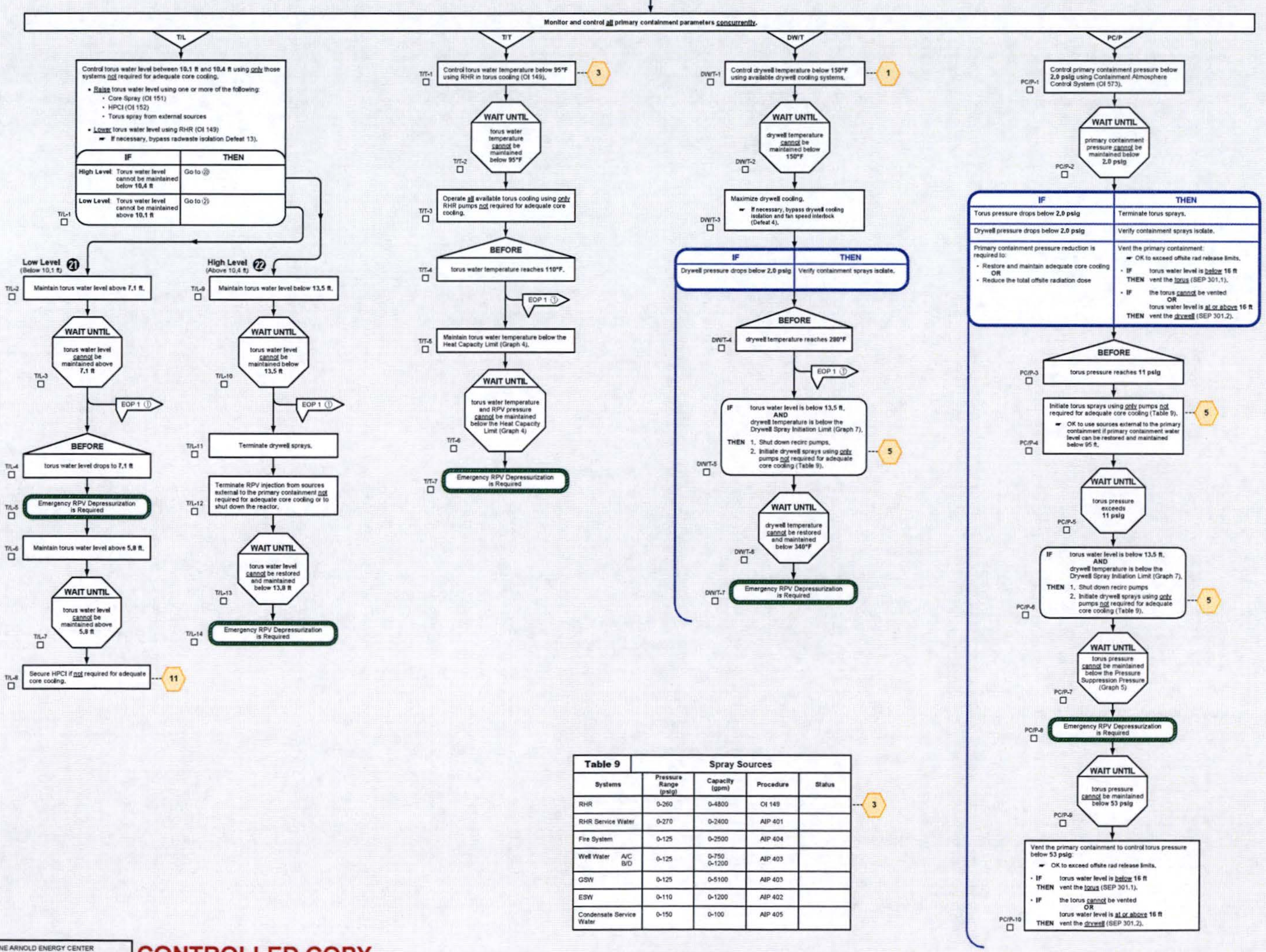
Torus water level above **10.4 ft**

Average torus water temperature above **95°F**

Average drywell temperature above **150°F**

Drywell pressure above **2.0 psig**

PC-1 ☐ Refer to EPIP 1.1 for EAL assessment.



Monitor and control all primary containment parameters concurrently.

T/L

T/L-1 ☐

Control torus water level between **10.1 ft** and **10.4 ft** using only those systems not required for adequate core cooling.

- Raise torus water level using one or more of the following:
  - Core Spray (OI 151)
  - HPCI (OI 152)
  - Torus spray from external sources
- Lower torus water level using RHR (OI 149)
  - If necessary, bypass radwaste isolation Defeat 13).

IF		THEN
High Level:	Torus water level cannot be maintained below **10.4 ft**	Go to ㉒
Low Level:	Torus water level cannot be maintained above **10.1 ft**	Go to ㉑

Low Level ㉑
(Below 10.1 ft)

T/L-2 ☐ Maintain torus water level above **7.1 ft**.

T/L-3 ☐ **WAIT UNTIL** torus water level cannot be maintained above **7.1 ft**

EOP 1 ①

T/L-4 ☐ **BEFORE** torus water level drops to **7.1 ft**

T/L-5 ☐ Emergency RPV Depressurization is Required

T/L-6 ☐ Maintain torus water level above **5.8 ft**.

T/L-7 ☐ **WAIT UNTIL** torus water level cannot be maintained above **5.8 ft**

T/L-8 ☐ Secure HPCI if not required for adequate core cooling. — 11

High Level ㉒
(Above 10.4 ft)

T/L-9 ☐ Maintain torus water level below **13.5 ft**.

T/L-10 ☐ **WAIT UNTIL** torus water level cannot be maintained below **13.5 ft**

EOP 1 ①

T/L-11 ☐ Terminate drywell sprays.

T/L-12 ☐ Terminate RPV injection from sources external to the primary containment not required for adequate core cooling or to shut down the reactor.

T/L-13 ☐ **WAIT UNTIL** torus water level cannot be restored and maintained below **13.8 ft**

T/L-14 ☐ Emergency RPV Depressurization is Required

T/T

T/T-1 ☐ Control torus water temperature below **95°F** using RHR in torus cooling (OI 149). — 3

T/T-2 ☐ **WAIT UNTIL** torus water temperature cannot be maintained below **95°F**

T/T-3 ☐ Operate all available torus cooling using only RHR pumps not required for adequate core cooling.

T/T-4 ☐ **BEFORE** torus water temperature reaches **110°F**.

EOP 1 ①

T/T-5 ☐ Maintain torus water temperature below the Heat Capacity Limit (Graph 4).

T/T-6 ☐ **WAIT UNTIL** torus water temperature and RPV pressure cannot be maintained below the Heat Capacity Limit (Graph 4)

T/T-7 ☐ Emergency RPV Depressurization is Required

DW/T

DW/T-1 ☐ Control drywell temperature below **150°F** using available drywell cooling systems. — 1

DW/T-2 ☐ **WAIT UNTIL** drywell temperature cannot be maintained below **150°F**

DW/T-3 ☐ Maximize drywell cooling.

- If necessary, bypass drywell cooling isolation and fan speed interlock (Defeat 4).

IF	THEN
Drywell pressure drops below **2.0 psig**	Verify containment sprays isolate.

DW/T-4 ☐ **BEFORE** drywell temperature reaches **280°F**

EOP 1 ①

DW/T-5 ☐
IF torus water level is below **13.5 ft**,
AND
drywell temperature is below the Drywell Spray Initiation Limit (Graph 7),
THEN
1. Shut down recirc pumps.
2. Initiate drywell sprays using only pumps not required for adequate core cooling (Table 9).

— 5

DW/T-6 ☐ **WAIT UNTIL** drywell temperature cannot be restored and maintained below **340°F**

DW/T-7 ☐ Emergency RPV Depressurization is Required

PC/P

PC/P-1 ☐ Control primary containment pressure below **2.0 psig** using Containment Atmosphere Control System (OI 573).

PC/P-2 ☐ **WAIT UNTIL** primary containment pressure cannot be maintained below **2.0 psig**

IF	THEN
Torus pressure drops below **2.0 psig**	Terminate torus sprays.
Drywell pressure drops below **2.0 psig**	Verify containment sprays isolate.
Primary containment pressure reduction is required to: • Restore and maintain adequate core cooling **OR** • Reduce the total offsite radiation dose	Vent the primary containment: OK to exceed offsite rad release limits. • **IF** torus water level is below **16 ft** **THEN** vent the torus (SEP 301.1). • **IF** the torus cannot be vented **OR** torus water level is at or above **16 ft** **THEN** vent the drywell (SEP 301.2).

PC/P-3 ☐ **BEFORE** torus pressure reaches **11 psig**

PC/P-4 ☐ Initiate torus sprays using only pumps not required for adequate core cooling (Table 9). — 5

- OK to use sources external to the primary containment if primary containment water level can be restored and maintained below 95 ft.

PC/P-5 ☐ **WAIT UNTIL** torus pressure exceeds **11 psig**

PC/P-6 ☐
IF torus water level is below **13.5 ft**,
AND
drywell temperature is below the Drywell Spray Initiation Limit (Graph 7),
THEN
1. Shut down recirc pumps
2. Initiate drywell sprays using only pumps not required for adequate core cooling (Table 9).

— 5

PC/P-7 ☐ **WAIT UNTIL** torus pressure cannot be maintained below the Pressure Suppression Pressure (Graph 5)

PC/P-8 ☐ Emergency RPV Depressurization is Required

PC/P-9 ☐ **WAIT UNTIL** torus pressure cannot be maintained below **53 psig**

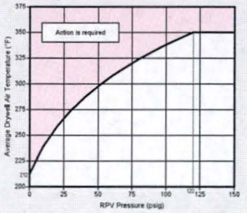
PC/P-10 ☐ Vent the primary containment to control torus pressure below **53 psig**:

- OK to exceed offsite rad release limits.
- **IF** torus water level is below **16 ft** **THEN** vent the torus (SEP 301.1).
- **IF** the torus cannot be vented **OR** torus water level is at or above **16 ft** **THEN** vent the drywell (SEP 301.2).

Table 9	Spray Sources				
Systems		**Pressure Range (psig)**	**Capacity (gpm)**	**Procedure**	**Status**
RHR		0-260	0-4800	OI 149	
RHR Service Water		0-270	0-2400	AIP 401	
Fire System		0-125	0-2500	AIP 404	
Well Water	A/C B/D	0-125	0-750 0-1200	AIP 403	
GSW		0-125	0-5100	AIP 403	
ESW		0-110	0-1200	AIP 402	
Condensate Service Water		0-150	0-100	AIP 405	

— 3

Graph 1: RPV Saturation Temperature



CAUTIONS

1 — The following restrictions apply to RPV level instruments:

1. If drywell air temperature is above the RPV Saturation Temperature (Graph 1), water in the instrument legs may boil. If boiling is suspected:
   a. Subtract 23 inches from Fuel Zone and Narrow Range GEMAC indications.
   b. Do not use Floodup and Wide Range Yarway instruments.
2. Floodup and Wide Range instruments may not be used below the Minimum Indicated Level for the indicated drywell temperature.

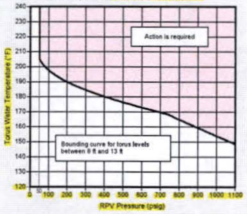
Level Instrument	Temperature Instrument	Drywell Temp (°F)	MIL (in.)
Wide Range Yarway LI-4539 (+8 to +218 in.)	TR-4383A Channel 1 (red)	100-150 151-200 201-250 251-300 301-350 Upscale	+9 +12 +16 +21 +25 +47
Wide Range Yarway LI-4540 (+8 to +218 in.)	TR-4383B Channel 2 (green)		
Floodup Range LI-4541 (+158 to +458 in.)	TR-4383A Channel 1 (red)	100-150 151-200 201-250 251-300 301-350 Upscale	+169 +175 +182 +190 +198 +256

3 — Operation of HPCI, RCIC, Core Spray, or RHR with suction from the torus and pump flow above the NPSH or vortex limit may damage equipment.

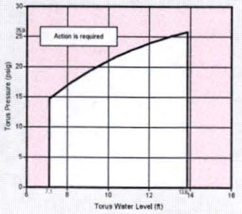
5 — Reducing primary containment pressure will reduce NPSH for pumps taking suction from the torus.

11 — Operation of HPCI with the turbine exhaust opening uncovered (5.8 ft) will increase torus pressure and may challenge primary containment limits.

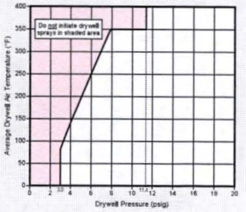
Graph 4: Heat Capacity Limit



Graph 5: Pressure Suppression Pressure



Graph 7: Drywell Spray Initiation Limit



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EOP 2 – PRIMARY CONTAINMENT CONTROL | REV. 17

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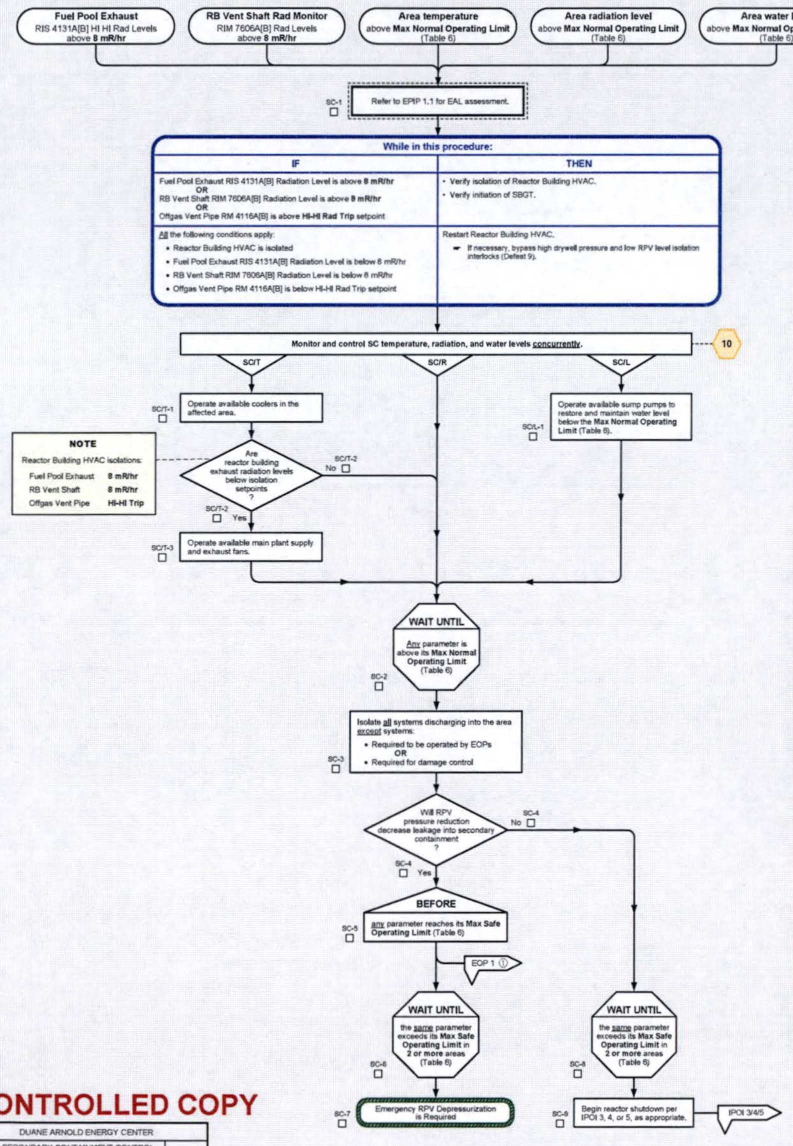
EOP 3
SECONDARY CONTAINMENT CONTROL



Fuel Pool Exhaust RIS 4131A[B] HI HI Rad Levels above **8 mR/hr**

RB Vent Shaft Rad Monitor RIM 7606A[B] Rad Levels above **8 mR/hr**

Area temperature above **Max Normal Operating Limit** (Table 6)

Area radiation level above **Max Normal Operating Limit** (Table 6)

Area water level above **Max Normal Operating Limit** (Table 6)

SC-1 ☐ Refer to EPIP 1.1 for EAL assessment.

While in this procedure:

IF	THEN
Fuel Pool Exhaust RIS 4131A[B] Radiation Level is above **8 mR/hr** **OR** RB Vent Shaft RIM 7606A[B] Radiation Level is above **8 mR/hr** **OR** Offgas Vent Pipe RM 4116A[B] is above **HI-HI Rad Trip** setpoint	• Verify isolation of Reactor Building HVAC. • Verify initiation of SBGT.
All the following conditions apply: • Reactor Building HVAC is isolated • Fuel Pool Exhaust RIS 4131A[B] Radiation Level is below 8 mR/hr • RB Vent Shaft RIM 7606A[B] Radiation Level is below 8 mR/hr • Offgas Vent Pipe RM 4116A[B] is below HI-HI Rad Trip setpoint	Restart Reactor Building HVAC. If necessary, bypass high drywell pressure and low RPV level isolation interlocks (Defeat 9).

Monitor and control SC temperature, radiation, and water levels concurrently. (10)

SC/T

SC/T-1 ☐ Operate available coolers in the affected area.

NOTE

Reactor Building HVAC isolations:

Fuel Pool Exhaust	**8 mR/hr**
RB Vent Shaft	**8 mR/hr**
Offgas Vent Pipe	**HI-HI Trip**

Are reactor building exhaust radiation levels below isolation setpoints ?

SC/T-2 No ☐

SC/T-2 ☐ Yes

SC/T-3 ☐ Operate available main plant supply and exhaust fans.

SC/R

SC/L

SC/L-1 ☐ Operate available sump pumps to restore and maintain water level below the **Max Normal Operating Limit** (Table 6).

WAIT UNTIL Any parameter is above its **Max Normal Operating Limit** (Table 6) SC-2 ☐

SC-3 ☐ Isolate all systems discharging into the area except systems:
- Required to be operated by EOPs
  OR
- Required for damage control

Will RPV pressure reduction decrease leakage into secondary containment ?

SC-4 No ☐

SC-4 ☐ Yes

BEFORE

SC-5 ☐ any parameter reaches its **Max Safe Operating Limit** (Table 6)

EOP 1 ①

WAIT UNTIL the same parameter exceeds its **Max Safe Operating Limit** in **2 or more areas** (Table 6) SC-6 ☐

SC-7 ☐ Emergency RPV Depressurization is Required

WAIT UNTIL the same parameter exceeds its **Max Safe Operating Limit** in **2 or more areas** (Table 6) SC-8 ☐

SC-9 ☐ Begin reactor shutdown per IPOI 3, 4, or 5, as appropriate.

IPOI 3/4/5

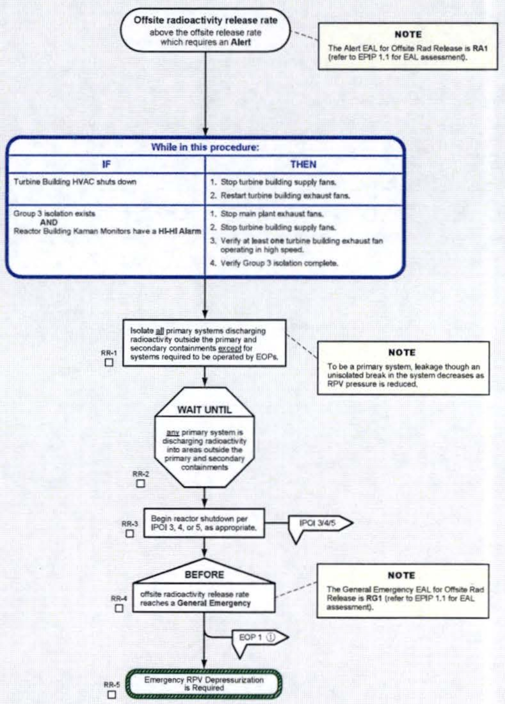
CAUTIONS

10 High spent fuel pool temperature or low spent fuel pool water level may affect conditions within secondary containment.

Table 6 **Secondary Containment Limits**

	Parameter and Areas		Max Normal Operating Limit	Max Safe Operating Limit	Value/Trend
Temperature	Area/Location	Indicator	°F	°F	
	A RHR—SE Corner Room Area				
	RHR SE CORNER ROOM AMBIENT	TR/TDR 2000A Ch 1	130	140	
	RHR SE CORNER ROOM DIFFERENTIAL	TR/TDR 2000A Ch 2	50	N/A	
	B RHR—NW Corner Room Area				
	RHR NW CORNER ROOM AMBIENT	TR/TDR 2000B Ch 1	130	140	
	RHR NW CORNER ROOM DIFFERENTIAL	TR/TDR 2000B Ch 2	50	N/A	
	HPCI Room Area				
	HPCI EMER COOLER AMBIENT	TR/TDR 2225A[B] Ch 1	175	310	
	HPCI ROOM AMBIENT	TR/TDR 2225A Ch 2	175	310	
	HPCI ROOM DIFFERENTIAL	TR/TDR 2225A[B] Ch 4[3]	50	N/A	
	RCIC Room Area				
	RCIC EMER COOLER AMBIENT	TR/TDR 2425A[B] Ch 1	175	300	
	RCIC ROOM AMBIENT	TR/TDR 2425A Ch 2	175	300	
	RCIC ROOM DIFFERENTIAL	TR/TDR 2425A[B] Ch 4	50	N/A	
	Torus Area				
	TORUS CATWALK NORTH AMBIENT	TR/TDR 2425A Ch 3	150	165	
	TORUS CATWALK WEST AMBIENT	TR/TDR 2425B Ch 2	150	165	
	TORUS CATWALK SOUTH AMBIENT	TR/TDR 2225A Ch 3	150	165	
	TORUS CATWALK EAST AMBIENT	TR/TDR 2225B Ch 2	150	165	
	TORUS CATWALK EAST DIFF	TR/TDR 2425A Ch 5	50	N/A	
	TORUS CATWALK WEST DIFF	TR/TDR 2425B Ch 5	50	N/A	
	TORUS CATWALK SOUTHWEST DIFF	TR/TDR 2225A Ch 5	50	N/A	
	TORUS CATWALK SOUTH DIFF	TR/TDR 2225B Ch 4	50	N/A	
	RB 786' South Area				
	RWCU PUMP ROOM AMBIENT	TR/TDR 2700A[B] Ch 1	130	212	
	RWCU HX ROOM AMBIENT	TR/TDR 2700A[B] Ch 2,3	130	212	
	RB 757' South Area				
	RWCU ABOVE TIP ROOM AMBIENT	TR/TDR 2700A[B] Ch 4,5	111.5	150	
	Steam Tunnel Area				
	STEAM TUNNEL AMBIENT	TR/TDR 2425B Ch 3	160	300	
	STEAM TUNNEL DIFFERENTIAL	TR/TDR 2225B Ch 5	70	N/A	
Radiation	Area/Location	Indicator	mR/hr	mR/hr	
	RB 757' South Area				
	RB RAILROAD ACCESS AREA	RI 9167	10	100	
	SOUTH CRD MODULE AREA	RI 9169	10	100	
	TIP ROOM	RI 9176	60	600	
	RB 757' North Area				
	NORTH CRD MODULE	RI 9168	10	100	
	CRD REPAIR ROOM	RI 9170	15	150	
	RB 786' North Area				
	RWCU SPENT RESIN ROOM	RI 9173	100	10^3	
	RWCU PHASE SEP TANK ROOM	RI 9177	20	200	
	RB 786' South Area				
	RWCU PUMP ROOM	RI 9156	10^3	10^4	
	RCWU HX ROOM	RI 9157	10^3	10^4	
	RB 812' North Area				
	MAIN PLANT EXHAUST FAN ROOM	RI 9171	60	600	
	JUNGLE ROOM	RI 9155	60	600	
	Refuel Floor Area				
	NEW FUEL VAULT AREA	RI 9153	10	100	
	NORTH REFUEL FLOOR	RI 9163	10	100	
	SOUTH REFUEL FLOOR	RI 9164	10	100	
	SPENT FUEL POOL AREA	RI 9178	100	10^3	
Water Level	Area/Location	Indicator	inches	inches	
	HPCI ROOM	LI 3768	2	6	
	RCIC ROOM	LI 3769	3	6	
	"A" RHR & CS SECR	LI 3770	2	10	
	"B" RHR & CS NWCR	LI 3771	2	10	
	TORUS AREA	LI 3772	2	12	

EOP 4
RADIOACTIVITY RELEASE CONTROL



Offsite radioactivity release rate above the offsite release rate which requires an **Alert**

NOTE

The Alert EAL for Offsite Rad Release is **RA1** (refer to EPIP 1.1 for EAL assessment).

While in this procedure:

IF	THEN
Turbine Building HVAC shuts down	1. Stop turbine building supply fans. 2. Restart turbine building exhaust fans.
Group 3 isolation exists **AND** Reactor Building Kaman Monitors have a **HI-HI Alarm**	1. Stop main plant exhaust fans. 2. Stop turbine building supply fans. 3. Verify **at least one** turbine building exhaust fan operating in high speed. 4. Verify Group 3 isolation complete.

RR-1 ☐ Isolate all primary systems discharging radioactivity outside the primary and secondary containments except for systems required to be operated by EOPs.

NOTE

To be a primary system, leakage though an unisolated break in the system decreases as RPV pressure is reduced.

WAIT UNTIL any primary system is discharging radioactivity into areas outside the primary and secondary containments RR-2 ☐

RR-3 ☐ Begin reactor shutdown per IPOI 3, 4, or 5, as appropriate.

IPOI 3/4/5

BEFORE

RR-4 ☐ offsite radioactivity release rate reaches a **General Emergency**

NOTE

The General Emergency EAL for Offsite Rad Release is **RG1** (refer to EPIP 1.1 for EAL assessment).

EOP 1 ①

RR-5 ☐ Emergency RPV Depressurization is Required

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DUANE ARNOLD ENERGY CENTER	
EOP 3 – SECONDARY CONTAINMENT CONTROL EOP 4 – RADIOACTIVITY RELEASE CONTROL	REV. 21

PROBABLE ANNUNCIATORS

None

PROBABLE INDICATIONS

1C35

- The amber DESIGN BASIS EARTHQUAKE (DBE) light is ON.
- The amber OPERATING BASIS EARTHQUAKE (OBE) light is ON.
- The amber .01G RECORDERS RUNNING light is ON.
- The white CONTINUITY light is OFF.
- The Seismic Wailing Alarm is sounding.
- Building vibration.

A Cooling Tower Valve House

- No power indicating light is operable.

INFORMATION

Earthquake	OBE	DBE
Ground Acceleration	0.06g	0.12g

CONDITIONAL STATEMENTS

IF while performing this procedure:

IF	Control Room access is regained **AND** personnel are available	**THEN**	when directed by the Emergency Response and Recovery Director resume control of unaffected components from the Control Room	_______
			AND	
			maintain control of Division II components from 1C388 until operability of Control Room instruments, indications and controls has been verified.	_______

NOTE

____

- Operations personnel evacuate to the Remote Shutdown Panel except: the STA, Shift Communicator, and on-site personnel not on shift evacuate to the TSC.
- The preferred evacuation route to the Remote Shutdown Panel is out the back door of the Control Room, and down the stairs. Emergency lighting is provided for this path.
- The alternate evacuation route to the Remote Shutdown Panel is out the front door of the Control Room, and down the stairs to access control. Emergency lighting is provided for this path.
- Since fire induced failure in 1C05 could adversely affect manual scram circuits, the initiation of ATWS ARI/RPT provides a redundant and diverse means of control rod insertion.

CAUTION

____

For Control Room evacuation as the result of a fire, transfer of control at panels 1C388, 1C389, 1C390, 1C391, 1C392 **is required to be completed within 20 minutes**.

……

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ 0.2 µCi/gm.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity > 0.2 µCi/gm and ≤ 2.0 µCi/gm DOSE EQUIVALENT I-131.	-------------NOTE----------- LCO 3.0.4.c is applicable. ----------------------------	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met. OR Reactor Coolant specific activity > 2.0 µCi/gm DOSE EQUIVALENT I-131.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	B.2.1 Isolate all main steam lines.	12 hours
	OR	
		(continued)

2.0 uci/gm chosen as EAL threashold since levels above that activity directly influence continued plant operation.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. ≤ 5 gpm unidentified LEAKAGE;

b. ≤ 25 gpm total LEAKAGE averaged over the previous 24 hour period; and

c. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce unidentified LEAKAGE increase to within limits. OR	4 hours (continued)

Developer Notes:

For EAL #1 leak rate value, entered the higher of 10 gpm or the value specified in DAEC's Technical Specifications for this type of leakage.

- **5 gpm per DAEC Tech Specs, so 10 gpm used in EAL**

For EAL #2 enter the higher of 25 gpm or the value specified in DAEC's Technical Specifications for this type of leakage.

- **DAEC uses a total leakage (identified + unidentified) spec of 25 gpm averaged over 24 hour period, so 25 gpm used in the EAL**

(9) During the approach to criticality, the operator withdrawing control rods should pause long enough between control rod notches to allow neutron count rate and period to stabilize, thus allowing a slow and controlled approach to the critical condition.

(10) When a control rod reaches position 48, perform a coupling check by attempting to withdraw the rod past position 48. If uncoupling should occur, stop control rod withdrawal, notify the CRS, and perform ARP 1C05A, D-7 (ROD OVERTRAVEL OUT).

(11) If criticality occurs significantly earlier or later than expected, notify the CRS.

(12) Approach the power range on a stable period of about 60-150 seconds. Do not achieve a sustained period of less than 50 seconds. If the period becomes too short, insert the notch and monitor for subcriticality.

(13) Each operable IRM channel must be indicating at least 5/40 scale on Range 1 prior to SRM count rate exceeding 10^6 cps with SRMs fully inserted. One IRM recorder on each RPS System should be in second speed (30 s/div) during startups while in the IRM Range. However, during extended stable operation in the IRM Range, it is permissible to shift the recorders to normal speed (30 min/div).

(14) Reactor plant heatup with MO-4629 and MO-4630, A/B RECIRC PUMP DISCH BYP in the closed position may cause bonnet over pressurization, resulting in failure of the valve to open due to pressure lock and damage to valve internals.

(15) Do not establish a vacuum in the main condenser until:

    (a) Steam seals are in operation.

    (b) Turbine is on turning gear.

    (c) Lube Oil Temperature > 80°F.

(16) Do not exceed a reactor pressure of 400 psig unless a reactor feed pump is in operation or the MSIVs are closed and the RCIC or HPCI Systems are operating.

(17) Do not retract IRMs until the MODE SWITCH is in RUN.

(18) Do not operate the mechanical vacuum pump above 10% reactor power to minimize the possibility of a hydrogen explosion or an untreated radioactivity release.

(19) Place the MODE SWITCH in RUN prior to reaching 12% reactor power.

BREAKPOINTS FOR REACTOR LEVEL CONTROL
Page 2 of 2

RPV Level (inches)	Item of Interest	Significance
-25	Minimum Steam Cooling RPV Water Level (MSCRWL)	• No guarantee that fuel cladding temperature can be kept <1500 °F • ED required in EOP 1 before -25 inches • SAG Entry in EOP 1 if cannot restore and maintain level above -25 inches and spray cooling cannot be established • Lower end of level control band in ATWS level/power control • Loss of ACC in ATWS Steam Cooling & SAG Entry
-39	Elevation of top of Jet Pump Suction (~2/3 Core Height)	• RPV water level following DBA LOCA • SAG Entry in EOP 1 if cannot restore and maintain level above -39 inches while spray cooling

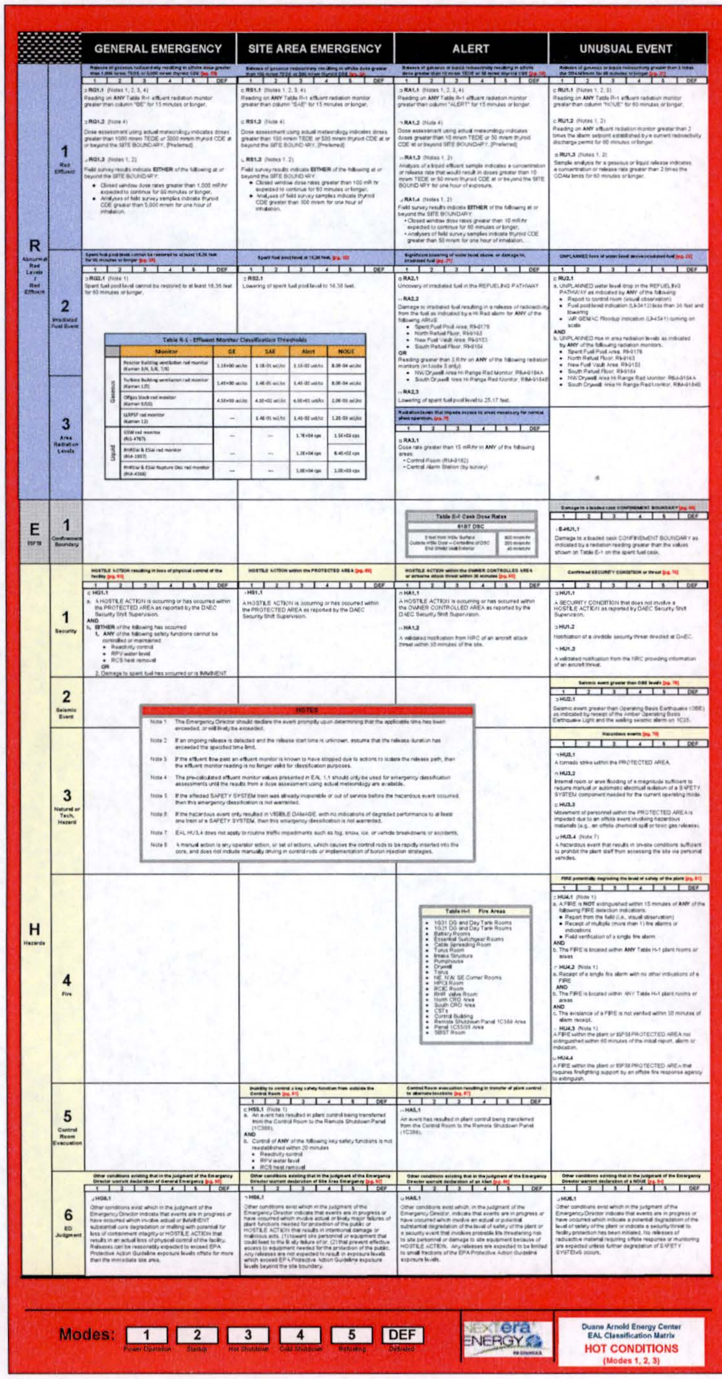
ATTACHMENT 5

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO
LICENSE AMENDMENT REQUEST TSCR-166

UPDATED DAEC EAL SCHEME WALLBOARDS
[FOR INFORMATION ONLY]

2 pages follow



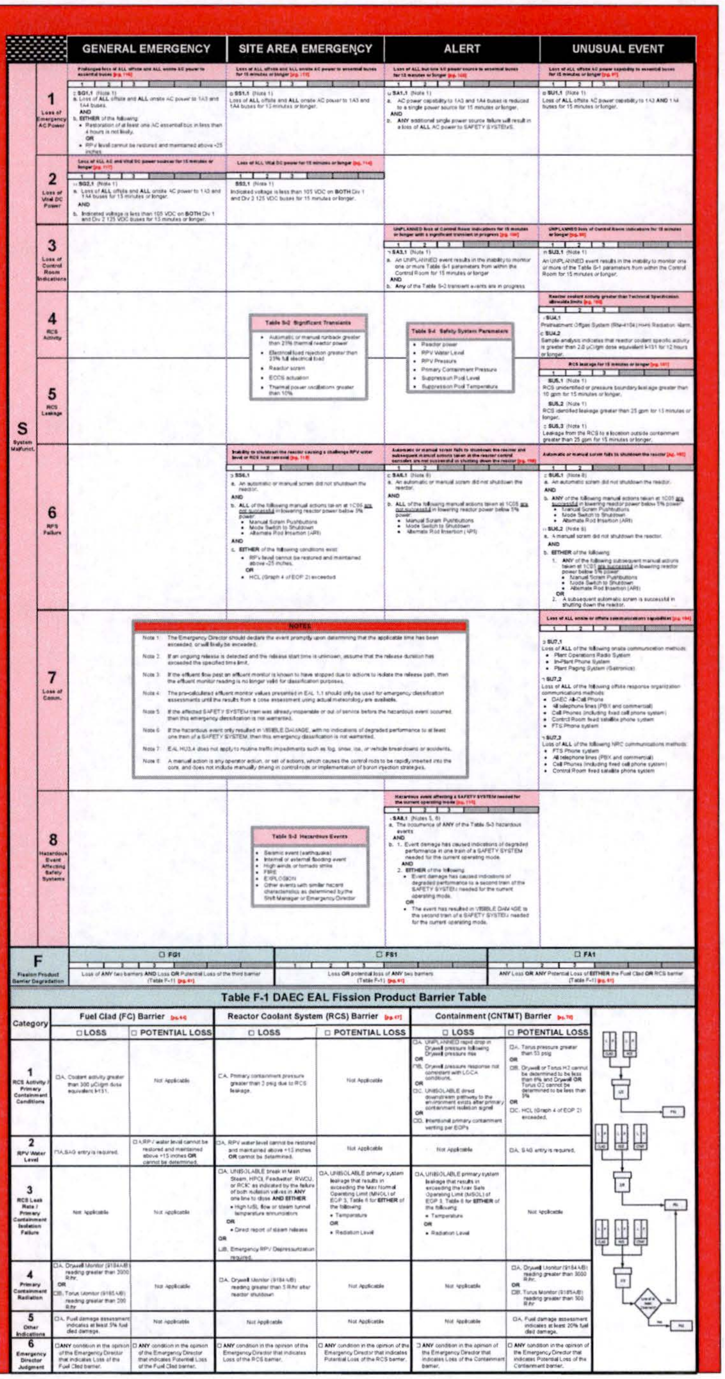
		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
R Abnormal Rad Levels / Rad Effluent	**1** Rad Effluent	RG1.1 (Notes 1, 2, 3, 4) Reading on **ANY** Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer. RG1.2 (Note 4) Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred] RG1.3 (Notes 1, 2) Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.	RS1.1 (Notes 1, 2, 3, 4) Reading on **ANY** Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer. RS1.2 (Note 4) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred] RS1.3 (Notes 1, 2) Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.	[illegible]	[illegible]
	2 Irradiated Fuel Event	RG2.1 (Note 1) Spent fuel pool level cannot be restored to at least 16.38 feet for 60 minutes or longer.	RS2.1 Lowering of spent fuel pool level to 16.38 feet.	[illegible]	[illegible]
	3 Area Radiation Levels			[illegible]	[illegible]
E ISFSI	**1** Confinement Boundary				[illegible]
H Hazards	**1** Security	[illegible]	[illegible]	[illegible]	[illegible]
	2 Seismic Event				[illegible]
	3 Natural or Tech. Hazard				[illegible]
	4 Fire				[illegible]
	5 Control Room Evacuation		[illegible]	[illegible]	
	6 ED Judgment	HG6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	HS6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	HA6.1 Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	HU6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	[illegible]	[illegible]	[illegible]	[illegible]
	Turbine Building ventilation rad monitor (Kaman 1/2)	[illegible]	[illegible]	[illegible]	[illegible]
	Offgas Stack rad monitor (Kaman 9/10)	[illegible]	[illegible]	[illegible]	[illegible]

Modes: 1 Power Operation · 2 Startup · 3 Hot Shutdown · 4 Cold Shutdown · 5 Refueling · DEF Defueled

Duane Arnold Energy Center
EAL Classification Matrix
HOT CONDITIONS
(Modes 1, 2, 3)



		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
S System Malfunct.	**1** Loss of Emergency AC Power	[illegible]	[illegible]	[illegible]	[illegible]
	2 Loss of Vital DC Power	[illegible]	[illegible]		
	3 Loss of Control Room Indications			[illegible]	[illegible]
	4 RCS Activity				[illegible]
	5 RCS Leakage				[illegible]
	6 RPS Failure		[illegible]	[illegible]	[illegible]
	7 Loss of Comm.				[illegible]
	8 Hazardous Event Affecting Safety Systems			[illegible]	

F Fission Product Barrier Degradation	☐ FG1	☐ FS1	☐ FA1
	Loss of **ANY** two barriers **AND** Loss **OR** Potential Loss of the third barrier (Table F-1)	Loss **OR** potential loss of **ANY** two barriers (Table F-1)	**ANY** Loss **OR ANY** Potential Loss of **EITHER** the Fuel Clad **OR** RCS barrier (Table F-1)

Table F-1 DAEC EAL Fission Product Barrier Table

Category	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CNTMT) Barrier	
	☐ LOSS	☐ POTENTIAL LOSS	☐ LOSS	☐ POTENTIAL LOSS	☐ LOSS	☐ POTENTIAL LOSS
1 RCS Activity / Primary Containment Conditions	☐A. Coolant activity greater than 300 µCi/gm dose equivalent I-131.	Not Applicable	☐A. Primary containment pressure greater than 2 psig due to RCS leakage.	Not Applicable	☐A. UNPLANNED rapid drop in Drywell pressure following Drywell pressure rise OR ☐B. Drywell pressure response not consistent with LOCA conditions. OR ☐C. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal OR ☐D. Intentional primary containment venting per EOPs	☐A. Torus pressure greater than 53 psig OR ☐B. Drywell or Torus H2 cannot be determined to be less than 6% and Drywell OR Torus O2 cannot be determined to be less than 5% OR ☐C. HCL (Graph 4 of EOP 2) exceeded.
2 RPV Water Level	☐A. SAG entry is required.	[illegible]	[illegible]	Not Applicable	Not Applicable	☐A. SAG entry is required.
3 RCS Leak Rate / Primary Containment Isolation Failure	Not Applicable	Not Applicable	[illegible]	[illegible]	[illegible]	Not Applicable
4 Primary Containment Radiation	[illegible]	Not Applicable	[illegible]	Not Applicable	Not Applicable	[illegible]
5 Other Indications	[illegible]	Not Applicable	Not Applicable	Not Applicable	Not Applicable	[illegible]
6 Emergency Director Judgment	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

R Abnormal Rad Levels / Rad Effluent

1 Rad Effluent

GENERAL EMERGENCY – **Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RG1.1** (Notes 1, 2, 3, 4)
Reading on **ANY** Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer.

☐ **RG1.2** (Note 4)
Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]

☐ **RG1.3** (Notes 1, 2)
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.

SITE AREA EMERGENCY – **Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RS1.1** (Notes 1, 2, 3, 4)
Reading on **ANY** Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer.

☐ **RS1.2** (Note 4)
Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]

☐ **RS1.3** (Notes 1, 2)
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.

ALERT – **Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RA1.1** (Notes 1, 2, 3, 4)
Reading on **ANY** Table R-1 effluent radiation monitor greater than column "ALERT" for 15 minutes or longer.

☐ **RA1.2** (Note 4)
Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond SITE BOUNDARY. [Preferred]

☐ **RA1.3** (Notes 1, 2)
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.

☐ **RA1.4** (Notes 1, 2)
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

UNUSUAL EVENT – **Release of gaseous or liquid radioactivity greater than 2 times the ODAM limits for 60 minutes or longer** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RU1.1** (Notes 1, 2, 3)
Reading on **ANY** Table R-1 effluent radiation monitor greater than column "NOUE" for 60 minutes or longer.

☐ **RU1.2** (Notes 1, 2)
Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

☐ **RU1.3** (Notes 1, 2)
Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODAM limits for 60 minutes or longer.

2 Irradiated Fuel Event

GENERAL EMERGENCY – **Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RG2.1** (Note 1)
Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.

SITE AREA EMERGENCY – **Spent fuel pool level at 16.36 feet.** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RS2.1**
Lowering of spent fuel pool level to 16.36 feet.

ALERT – **Significant lowering of water level above, or damage to, irradiated fuel** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RA2.1**
Uncovery of irradiated fuel in the REFUELING PATHWAY

☐ **RA2.2**
Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by a Hi Rad alarm for **ANY** of the following ARMS:
- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164

OR

Reading greater than 3 R/hr on **ANY** of the following radiation monitors (in Mode 5 only):
- N/W Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

☐ **RA2.3**
Lowering of spent fuel pool level to 25.17 feet.

UNUSUAL EVENT – **UNPLANNED loss of water level above irradiated fuel** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RU2.1**
a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:
- Report to control room (visual observation)
- Fuel pool level indication (LI-3413) less than 36 feet and lowering
- WR GEMAC Floodup indication (LI-4541) coming on scale

AND

b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors:
- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164
- N/W Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uCi/cc	1.1E-01 uCi/cc	1.1E-03 uCi/cc	8.8E-04 uCi/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uCi/cc	1.4E-01 uCi/cc	1.4E-02 uCi/cc	8.8E-04 uCi/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uCi/cc	4.5E+02 uCi/cc	4.5E+01 uCi/cc	3.8E-01 uCi/cc
	LLRPSF rad monitor (Kaman 12)	—	1.4E-00 uCi/cc	1.4E-02 uCi/cc	1.2E-03 uCi/cc
Liquid	GSW rad monitor (RIS-4767)	—	—	1.3E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	—	—	1.3E+04 cps	8.4E+03 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	—	—	1.8E+04 cps	1.0E+03 cps

3 Area Radiation Levels

ALERT – **Radiation levels that impede access to areas necessary for normal plant operation.** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **RA3.1**
Dose rate greater than 15 mR/hr in **ANY** of the following areas:
- Control Room (RM-9162)
- Central Alarm Station (by survey)

E ISFSI

1 Confinement Boundary

Table E-1 Cask Dose Rates

#1BT DSC	
3 feet from HSM Surface	800 mrem/hr
Outside HSM Door – Centerline of DSC	200 mrem/hr
End Shield Wall Exterior	40 mrem/hr

UNUSUAL EVENT – **Damage to a loaded cask CONFINEMENT BOUNDARY** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **E-HU1.1**
Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by a radiation reading greater than the values shown on Table E-1 on the spent fuel cask.

H Hazards

1 Security

GENERAL EMERGENCY – **HOSTILE ACTION resulting in loss of physical control of the facility** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HG1.1**
a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.

AND

b. **EITHER** of the following has occurred:
1. **ANY** of the following safety functions cannot be controlled or maintained
   - Reactivity control
   - RPV water level
   - RCS heat removal

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

SITE AREA EMERGENCY – **HOSTILE ACTION within the PROTECTED AREA** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HS1.1**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.

ALERT – **HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HA1.1**
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the DAEC Security Shift Supervision.

☐ **HA1.2**
A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

UNUSUAL EVENT – **Confirmed SECURITY CONDITION or threat** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HU1.1**
A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by DAEC Security Shift Supervision.

☐ **HU1.2**
Notification of a credible security threat directed at DAEC.

☐ **HU1.3**
A validated notification from the NRC providing information of an aircraft threat.

2 Seismic Event

UNUSUAL EVENT – **Seismic event greater than OBE levels** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HU2.1**
Seismic event greater than Operating Basis Earthquake (OBE) as indicated by receipt of the Amber Operating Basis Earthquake Light and the wailing seismic alarm on 1C35.

3 Natural or Tech. Hazard

NOTES

Note 1: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 6: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

Note 7: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

UNUSUAL EVENT – **Hazardous events** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HU3.1**
A tornado strike within the PROTECTED AREA

☐ **HU3.2**
Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

☐ **HU3.3**
Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

☐ **HU3.4** (Note 7)
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

4 Fire

Table H-1 Fire Areas
- 1G31 DG and Day Tank Rooms
- 1G21 DG and Day Tank Rooms
- Battery Rooms
- Essential Switchgear Rooms
- Cable Spreading Room
- Torus Room
- Intake Structure
- Pumphouse
- Drywell
- Torus
- NE, NW, SE Corner Rooms
- HPCI Room
- RCIC Room
- RHR Valve Room
- North CRD Area
- South CRD Area
- CSTs
- Control Building
- Remote Shutdown Panel 1C388 Area
- Panel 1C55/56 Area
- SBGT Room

UNUSUAL EVENT – **FIRE potentially degrading the level of safety of the plant** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HU4.1**
a. A FIRE is **NOT** extinguished within 15 min. of **ANY** of the following FIRE detection indications:
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

b. The FIRE is located within **ANY** Table H-1 plant rooms or areas

☐ **HU4.2**
a. Receipt of a single fire alarm with no other indications of a FIRE

AND

b. The FIRE is located within ANY Table H-1 plant rooms or areas

AND

c. The existence of a FIRE is not verified within 30 minutes of alarm receipt.

☐ **HU4.3**
A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60 minutes of the initial report, alarm or indication.

☐ **HU4.4**
A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

5 Control Room Evacuation

SITE AREA EMERGENCY – **Inability to control a key safety function from outside the Control Room** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HS5.1**
a. An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).

AND

b. Control of **ANY** of the following key safety functions is not reestablished within 20 minutes:
- Reactivity control
- RPV water level
- RCS heat removal

ALERT – **Control Room evacuation resulting in transfer of plant control to alternate locations** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HA5.1**
An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).

6 ED Judgment

GENERAL EMERGENCY – **Other conditions existing that in the judgment of the Emergency Director warrant declaration of General Emergency** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HG6.1**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY – **Other conditions existing that in the judgment of the Emergency Director warrant declaration of Site Area Emergency** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HS6.1**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

ALERT – **Other conditions existing that in the judgment of the Emergency Director warrant declaration of an Alert** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HA6.1**
Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

UNUSUAL EVENT – **Other conditions existing that in the judgment of the Emergency Director warrant declaration of a NOUE** [pg. [illegible]]

1 | 2 | 3 | 4 | 5 | DEF

☐ **HU6.1**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Modes: 1 Power Operation | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | DEF Defueled

Duane Arnold Energy Center
EAL Classification Matrix

COLD CONDITIONS
(Modes 4, 5, and Defueled)

NEXTera ENERGY RESOURCES

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

C Cold SD/ Refueling System Malfunct.

1 RCS Level

GENERAL EMERGENCY – **Loss of RPV inventory affecting fuel clad integrity with containment challenged** [pg. [illegible]]

4 | 5

☐ **CG1.1** (Note 1)
a. RPV level less than +15 inches for 30 minutes or longer.

AND

b. **ANY** indication from the Secondary Containment Challenge Table C-1.

☐ **CG1.2** (Note 1)
a. RPV level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **EITHER** of the following:
- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr.
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicated core uncovery.

AND

c. **ANY** indication from the Secondary Containment Challenge Table C-1.

SITE AREA EMERGENCY – **Loss of RPV inventory affecting core decay heat removal capability** [pg. [illegible]]

4 | 5

☐ **CS1.1**
a. CONTAINMENT CLOSURE not established.

AND

b. RPV level less than +64 inches.

☐ **CS1.2**
a. CONTAINMENT CLOSURE established.

AND

b. RPV level less than +15 inches.

☐ **CS1.3** (Note 1)
a. RPV level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **EITHER** of the following:
- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr.
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery.

ALERT – **Loss of RPV inventory** [pg. [illegible]]

4 | 5

☐ **CA1.1**
Loss of RPV inventory as indicated by level less than 119.5 inches.

☐ **CA1.2** (Note 1)
a. RPV level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool due to a loss of RPV inventory

UNUSUAL EVENT – **UNPLANNED loss of RPV inventory for 15 minutes or longer** [pg. [illegible]]

4 | 5

☐ **CU1.1** (Note 1)
UNPLANNED loss of reactor coolant results in RPV level less than a required lower limit for 15 minutes or longer.

☐ **CU1.2**
a. RPV level cannot be monitored

AND

b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool.

Table C-1 Secondary Containment Challenge
- CONTAINMENT CLOSURE not established *
- Drywell Hydrogen or Torus Hydrogen greater than 6% **AND** Drywell Oxygen or Torus Oxygen greater than 5%
- UNPLANNED increase in containment pressure
- Secondary containment radiation monitors above max safe operating limits (MSOL) of EOP 3, Table 6

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

2 Loss of Emergency AC Power

ALERT – **Loss of all offsite and all onsite AC power to essential buses for 15 minutes or longer.** [pg. [illegible]]

4 | 5 | DEF

☐ **CA2.1** (Note 1)
Loss of **ALL** offsite and **ALL** onsite AC Power to 1A3 and 1A4 buses for 15 minutes or longer.

UNUSUAL EVENT – **Loss of all but one AC power source to essential buses for 15 minutes or longer** [pg. [illegible]]

4 | 5 | DEF

☐ **CU2.1** (Note 1)
a. AC power capability to 1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.

AND

b. Any additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.

3 RCS Temp.

Table C-2 RCS Heat-up Duration Thresholds

RCS Integrity	Containment Closure Status	Heat-up Duration
Intact	Not applicable	60 minutes*
Not intact	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.

ALERT – **Inability to maintain the plant in cold shutdown** [pg. [illegible]]

4 | 5

☐ **CA3.1** (Note 1)
UNPLANNED increase in RCS temperature to greater than 212°F for greater than the duration specified in Table C-2.

☐ **CA3.2**
UNPLANNED RCS pressure increase greater than 10 psig due to a loss of RCS cooling.

UNUSUAL EVENT – **UNPLANNED increase in RCS temperature** [pg. [illegible]]

4 | 5

☐ **CU3.1**
UNPLANNED increase in RCS temperature to greater than 212°F

☐ **CU3.2** (Note 1)
Loss of **ALL** RCS temperature and RPV level indication for 15 minutes or longer.

4 Loss of Vital DC Power

UNUSUAL EVENT – **Loss of Vital DC power for 15 minutes or longer** [pg. [illegible]]

4 | 5

☐ **CU4.1** (Note 1)
Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

5 Loss of Comm.

NOTES

Note 1: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 6: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

Note 7: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

UNUSUAL EVENT – **Loss of ALL onsite or offsite communications capabilities** [pg. [illegible]]

4 | 5 | DEF

☐ **CU5.1**
Loss of **ALL** of the following onsite communication methods:
- Plant Operations Radio System
- In-Plant Phone System
- Plant Paging System (Gaitronics)

☐ **CU5.2**
Loss of **ALL** of the following offsite response organization communications methods:
- DAEC All-Call phone
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- FTS Phone system

☐ **CU5.3**
Loss of **ALL** of the following NRC communications methods:
- FTS Phone system
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system

6 Hazardous Event Affecting Safety Systems

Table C-3 Hazardous Events
- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

ALERT – **Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode** [pg. [illegible]]

4 | 5

☐ **CA6.1** (Notes 5, 6)
a. The occurrence of **ANY** of the Table C-3 hazardous events

AND

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

   AND

   2. **EITHER** of the following:
   - Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode.

     OR
   - The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.