ATTACHMENT 2

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130 pages follow

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

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TABLE OF CONTENTS

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.

<table>
<tr><td colspan="4">Emergency Classification Level</td></tr>
<tr><td>Unusual Event</td><td>Alert</td><td>SAE</td><td>GE</td></tr>
<tr><td>↓</td><td>↓</td><td>↓</td><td>↓</td></tr>
<tr><td>Initiating Condition</td><td>Initiating Condition</td><td>Initiating Condition</td><td>Initiating Condition</td></tr>
<tr><td>↓</td><td>↓</td><td>↓</td><td>↓</td></tr>
<tr><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td></tr>
<tr><td colspan="4">(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.</td></tr>
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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/trip 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.

For EAL thresholds that specify a duration of the off-normal condition, the NRC expects that the emergency declaration process run concurrently with the specified threshold duration. Once the off-normal condition has existed for the duration specified in the EAL, no further effort on this declaration is necessary—the EAL has been exceeded. Consider as an example, the EAL "fire which is not extinguished within 15 minutes of detection." On receipt of a fire alarm, the plant fire brigade is dispatched to the scene to begin fire suppression efforts.

- If the fire brigade reports that the fire can be extinguished before the specified duration, the emergency declaration is placed on hold while firefighting activities continue. If the fire brigade is successful in extinguishing the fire within the specified duration from detection, no emergency declaration is warranted based on that EAL.

- If the fire is still burning after the specified duration has elapsed, the EAL is exceeded, no further assessment is necessary, and the emergency declaration would be made promptly. As used here, "promptly" means at the first available opportunity (e.g., if the Shift Manager is receiving an update from the fire brigade at the 15-minute mark, it is expected that the declaration will occur as the next action after the call ends).

- If, for example, the fire brigade notifies the shift supervision 5 minutes after detection that the brigade itself cannot extinguish the fire such that the EAL will be met imminently and cannot be avoided, the NRC would not consider it a violation of the licensee's emergency plan to declare the event before the EAL is met (e.g., the 15-minute duration has elapsed). While a prompt declaration would be beneficial to public health and safety and is encouraged, it is not required by regulation.

- In all of the above, the fire duration is measured from the time the alarm, indication, or report was first received by the plant operators. Validation or confirmation establishes that the fire started as early as the time of the alarm, indication, or report.

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. For example:

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

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

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

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

5.4 CONSIDERATION OF MODE CHANGES DURING CLASSIFICATION

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

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if 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/trip the reactor followed by a successful manual scram/trip or an earthquake.

5.8 CLASSIFICATION OF TRANSIENT CONDITIONS

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

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

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

> 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

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

RU1.1 Reading on **ANY** of the following effluent radiation monitors greater than the reading shown for 60 minutes or longer:

Effluent Monitor Classification Thresholds		
	Monitor	**NOUE**
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.0E-03 uCi/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.0E-03 uCi/cc
	Offgas Stack rad monitor (Kaman 9/10)	2.0E-01 uCi/cc
	LLRPSF rad monitor (Kaman 12)	1.0E-03 uCi/cc
Liquid	GSW rad monitor (RIS-4767)	2.0E+03 CPS
	RHRSW & ESW rad monitor (RM-1997)	8.0E+02 CPS
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	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 analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC 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.

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 Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RA1.1 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** of the following radiation monitors greater than the reading shown for 15 minutes or longer:

Effluent Monitor Classification Thresholds		
	Monitor	**Alert**
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.0E-02 uCi/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.0E-02 uCi/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+01 uCi/cc
	LLRPSF rad monitor (Kaman 12)	1.0E-02 uCi/cc
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

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 ft.

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 ARM (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 Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RS1.1 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** of the following radiation monitor greater than the reading shown for 15 minutes or longer:

Effluent Monitor Classification Thresholds

	Monitor	SAE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.0E-01 uCi/cc
Gaseous	Turbine Building ventilation rad monitor (Kaman 1/2)	1.0E-01 uCi/cc
Gaseous	Offgas Stack rad monitor (Kaman 9/10)	4.5E+02 uCi/cc
Gaseous	LLRPSF rad monitor (Kaman 12)	1.0E-01 uCi/cc

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.

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 General Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RG1.1 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** of the following radiation monitors greater than the reading shown for 15 minutes or longer:

Effluent Monitor Classification Thresholds		
	Monitor	**GE**
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.0E+00 uCi/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.0E+00 uCi/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uCi/cc

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.

RG2

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 General Emergency promptly upon determining that 60 minutes 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 Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU1.1 UNPLANNED loss of reactor coolant results in RPV level less than **ANY** of the following for 15 minutes or longer:

a. In Mode 4, RPV water level less than 170"

OR

b. In Mode 5, if RPV level band is established above the RPV flange and RPV water level drops below the RPV flange.

OR

c. In Mode 5, if RPV level band is established below the RPV flange and RPV water level drops below RPV level band.

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 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.

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 Unusual 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. Systems 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.

CU3

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 Unusual Event promptly upon determining that 15 minutes 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 Unusual Event promptly upon determining that 15 minutes 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. Systems 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.

Minimum DC bus voltage selected due to automatic trip of the inverters at 105 VDC decreasing.

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

CU5

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 Alert promptly upon determining that 15 minutes 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 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.

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 Alert promptly upon determining that 15 minutes 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. Systems 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 Alert 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 the following table.

Table: 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.

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 following hazardous events:
- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- River level above 757 feet
- River Water Supply (RWS) pit low level alarm
- 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,

OR

- Loss of the safety function of a single train SAFETY SYSTEM.

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. Systems 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.

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 Site Area Emergency promptly upon determining that 30 minutes 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 **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

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 RCS/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 RCS/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 General Emergency promptly upon determining that 30 minutes 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 (see below).

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

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

AND

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

Secondary Containment Challenge Table
• 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 RCS/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 an on-contact radiation reading greater than the values shown below on the surface of the spent fuel cask.

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

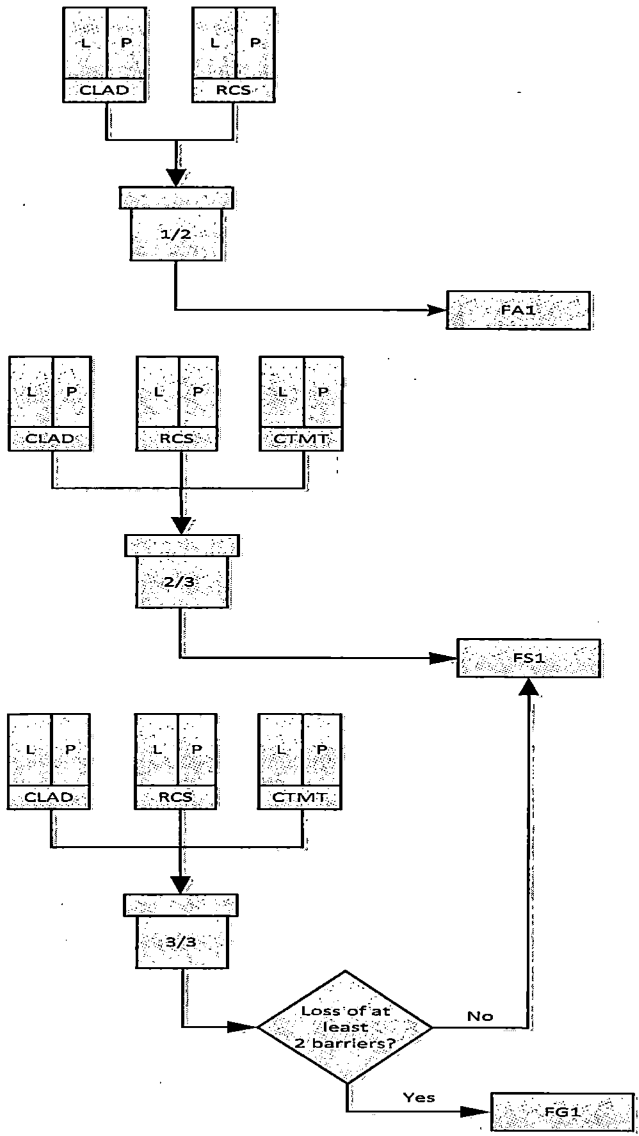

Table 9-F: DAEC EAL Fission Product Barrier Table

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. Primary Containment Conditions		**1. Primary Containment Conditions**		**1. Primary Containment Conditions**	
Not Applicable	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. RCS Leak Rate		**3. RCS Leak Rate**		**3. RCS Leak Rate**	
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 200 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 9-F

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. Primary Containment Conditions

There is no Loss or Potential Loss threshold associated with Primary Containment Condition.

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.

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 SA5 or SS5 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. RCS Leak Rate

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

4. Primary Containment Radiation

<u>Loss 4.A</u>

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

<u>Loss 4.B</u>

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

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

DAEC FUEL CLAD BARRIER THRESHOLDS (cont.):

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.

Developer Notes:

Loss and/or Potential Loss 5.A

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

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

6. Emergency Director Judgment

Loss 6.A

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

Potential Loss 6.A

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

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.

DAEC RCS BARRIER THRESHOLDS (cont.):

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.

Developer Notes:

Loss and/or Potential Loss 5.A

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

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

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the 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 HCTL is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

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. RCS Leak Rate

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

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 $\pm$ 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.

HU3.5 River level above 757 feet.

HU3.6 River Water Supply (RWS) pit low level alarm.

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. Systems 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.

EAL HU3.5 addresses the observed effects of flooding in accordance with AOP 902 (Flood). Plant site finished grade is at elevation 757.0 feet. Personnel doors and railroad and truck openings at or near grade would require protection in the event of a flood above elevation 757.0 feet. Therefore, EAL 6 uses a threshold of flood water levels above 757.0 feet. .

EAL HU3.6 addresses the effects of loss of river water make-up capability. The intake structure for the safety-related water supply systems (river water, RHR service water, and emergency service water) is located on the west bank of the Cedar River. River levels below the intake structure inlet or a blockage of the intake would result in a loss of the ability to provide make-up water for safety-related systems. The overflow weir is at elevation 724 feet 6 inches. River level at or below this elevation will result in all river flow being diverted to the safety related water supply systems. The top of the intake structure around the pump wells is at elevation 724 feet. If the river water level dropped to this level, the pump suction would have no continuous supply. Blockages of the intake structure may result from debris, ice, or aquatic life. A loss of flow into the intake structure, due to a blockage or low river level, will result in the pit level lowering to the alarm setpoint (723.0 feet) and a resulting alarm in the Control Room. Therefore, this EAL uses a threshold of low pit level as a potential substantial degradation of the ultimate heat sink capability.

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 Unusual 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 Safe Shutdown/Vital Areas	
Category	**Area**
Electrical Power	1G31 DG and Day Tank Rooms, 1G21 DG and Day Tank Rooms, Battery Rooms, Essential Switchgear Rooms, Cable Spreading Room
Heat Sink / Coolant Supply	Torus Room, Intake Structure, Pumphouse
Containment	Drywell, Torus
Emergency Systems	NE, NW, SE Corner Rooms, HPCI Room, RCIC Room, RHR Valve Room, North CRD Area, South CRD Area, CSTs
Other	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.

<u>EAL HU4.1</u>

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.

<u>EAL HU4.2</u>

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. Systems 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 Site Area Emergency promptly upon determining that 20 minutes 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 Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SU1.1 Loss of **ALL** offsite AC power capability to 1A3 **AND** 1A4 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 Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SU3.1 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.

- 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. Systems 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 Unusual Event promptly upon determining that 15 minutes 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 trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor 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 Alert promptly upon determining that 15 minutes 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. Systems 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.

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 Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SA3.1 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.

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

AND

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
- 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. Systems 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. <u>ALL</u> of the following manual actions taken at 1C05 <u>are not successful</u> 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).

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.
- For a single train SAFETY SYSTEM, degraded performance which results in loss of the safety function of the SAFETY SYSTEM

SA8.1 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
- River level above 757 feet
- River Water Supply (RWS) pit low level alarm
- 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,

OR

- Loss of the safety function of a single train SAFETY SYSTEM.

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. Systems 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.

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 Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SS1.1 Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 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. Systems 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 Site Area Emergency promptly upon determining that 15 minutes 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. Systems 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.

Minimum DC bus voltage selected due to automatic trip of the inverters at 105 VDC decreasing.

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.

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 General Emergency promptly upon determining that 4 hours 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. Systems 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 General Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SG2.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 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. Systems 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.

Minimum DC bus voltage selected due to automatic trip of the inverters at 105 VDC decreasing.

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

[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

LICENSE AMENDMENT REQUEST TSCR-166

DEVIATIONS AND DIFFERENCES MATRIX

100 pages follow

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	Added text from Section IV.H.7 of NSIR/DPR-ISG-01 explaining how to treat concurrent time periods when making an emergency declaration. Information was added to address a frequently asked question by the DAEC operators.	V2
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** of the following effluent radiation monitors greater than the reading shown for 60 minutes or longer: [inserted Table 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.	V3
	(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	V4

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.	V5

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** of the following radiation monitors greater than the reading shown for 15 minutes or longer: [inserted Table of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V6, V7
	(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).	V8
	(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.	V9

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.	V10

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** of the following radiation monitors greater than the reading shown for 15 minutes or longer: [inserted Table of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V11
	(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	V12
	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.	V12

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** of the following radiation monitors greater than the reading shown for 15 minutes or longer: [inserted Table of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V13
	(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	V12
	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.	V12

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 **ANY** of the following for 15 minutes or longer: a. In Mode 4, RPV water level less than 170" **OR** b. In Mode 5, if RPV level band is established above the RPV flange and RPV water level drops below the RPV flange. **OR** c. In Mode 5, if RPV level band is established below the RPV flange and RPV water level drops below RPV level band.	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

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
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 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.	(1) a. AC power capability to 1A3 and 1A4 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.	Difference	Global Comment #9, 12, & 13 Intent and meaning of the EALs are not altered.	V14

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.	V15

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	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	V16 V17

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.	V16

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	V18
	(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.	V14

<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 the following table:</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="2">RCS Heat-up Duration Thresho</th></tr>
<tr><th>RCS Integrity</th><th>Containment Closure Status</th></tr>
<tr><td>Intact</td><td>Not Applicable</td></tr>
<tr><td rowspan="2">Not intact</td><td>Established</td></tr>
<tr><td>Not Established</td></tr>
<tr><td colspan="2">* If an RCS heat removal system is in operatior … emperature is being reduced, t</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>V19</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 following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • River level above 757 feet • River Water Supply (RWS) pit low level alarm • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director	Difference	Global Comment #9, 12 & 13	V20 V21

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. **OR** 1. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure 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** 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. • Loss of the safety function of a single train SAFETY SYSTEM.	Deviation Deviation	Adopted the revised EAL wording provided in approved EAL FAQ 2016-02. Adopted the revised EAL wording provided in approved EAL FAQ 2016-02; with the addition of a 3rd choice due to DAEC having single train SAFETY SYSTEM Intent and meaning of the EALs are not altered.	V22 V22

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	V23
	(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	V23

CS1 (cont.)	(3) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer.	(3) 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 • (Other site-specific indications)	**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 #9 &13	V24
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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	V23
	(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** 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	V24
	AND c. **ANY** indication from the Containment Challenge Table (see below).	**AND** c. **ANY** indication from Containment Challenge Table (see below)	Verbatim		None
	Containment Challenge Table ONTAINMENT CLOSURE not established* Explosive mixture) exists inside containment NPLANNED increase in containment pressure econdary containment radiation monitor reading ite specific value) [BWR]	**Containment Challenge Table C** • CONTAINMENT CLOSURE not established • Drywell Hydrogen or Torus Hydrogen gre **AND** Drywell Oxygen or Torus Oxygen gre • UNPLANNED increase in containment pre • Secondary containment radiation monito safe operating limits (MSOL) of EOP 3, Tal	Difference	Global Comment #9	V25 V26
	* 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

<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 an on-contact radiation reading greater than the values shown below on the surface of the spent fuel cask.

<table><tr><th colspan="2">61BT DSC</th></tr><tr><td>3 feet from HSM Surface</td><td>800 mrem/hr</td></tr><tr><td>Outside HSM Door – Centerline of DSC</td><td>200 mrem/hr</td></tr><tr><td>End Shield Wall Exterior</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>V27</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>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</td><td>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</td><td>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</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 9-F: 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 Renamed to **1. Primary Containment Conditions**	A. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).	Not Applicable	Not Applicable	Not Applicable	Difference	Renamed Category from "RCS" Activity" to "Primary Containment Conditions" to better align category with thresholds being assessed. Fuel Clad LOSS 1.A carried over to OTHER INDICATIONS category as LOSS 5.A
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 V28 General 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.	Not Applicable	Not Applicable	Difference	Renamed Category from "Not applicable" to "RCS Leak Rate" to better align category with thresholds being assessed.

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**		
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 200 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 200 R/hr	Not Applicable	Difference	V29 - V30 Global Comment #9 RCS @300 uci/cc readable on Drywell monitor, so that lower value was used versus value for loss of both RCS and Fuel clad barriers. Torus monitor value follows the developer guidance for *"instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS radioactivity concentration equal to 300 uCi/gm dose equivalent I-131, into the primary containment atmosphere*
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.
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.	Difference	Emergency Director changed to Emergency Director to align with site terminology.

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	V31 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	V23 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	V32 Global Comment #9 Added site-specific indication of an unisolable steam line break which includes failure of both isolation valves to LOSS 3.A.

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 V29
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.	Difference	Emergency Director changed to Emergency Director to align with site terminology.

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**		
1. Primary Containment Pressure Renamed to **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 V25 V33 V34 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. RCS Leak Rate	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 V35 Loss 3.A and 3.B moved to category 1 "Primary Containment Conditions"
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 V29

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.	Difference	Emergency Director changed to Emergency Director to align with site terminology.

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	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 #
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.	V36

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)	(5) River level above 757 feet.	Difference	Global Comment #9	V37
		(6) River Water Supply (RWS) pit low level alarm.	Difference	Global Comment #9 Added as a 6th EAL for this IC versus a list in EAL #5 to maintain consistent format Intent and meaning of the EALs are not altered.	V38

<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">HU4</td><td>Recognition Category: HU4</td><td>HU4</td><td>Verbatim</td><td>Global Comment #11, 14</td><td>None</td></tr>
<tr><td>Initiating Condition: FIRE potentially degrading the level of safety of the plant.</td><td>FIRE potentially degrading the level of safety of the plant.</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) 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</td><td>(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</td><td>Difference</td><td>Global Comment #12 & 13</td><td>None</td></tr>
<tr><td>b. The FIRE is located within ANY of the following plant rooms or areas:
(site-specific list of plant rooms or areas)</td><td>b. The FIRE is located within ANY Table H-1 plant rooms or areas.
<table>
<tr><th colspan="2">Table H-1 Safe Shutdown/Vital Areas</th></tr>
<tr><th>Category</th><th>Area</th></tr>
<tr><td>Electrical Power</td><td>1G31 DG and Day Tank Rooms, 1G21 DG and Day Tank Rooms, Battery Rooms, Essential Switchgear Rooms, Cable Spreading Room</td></tr>
<tr><td>Heat Sink / Coolant Supply</td><td>Torus Room, Intake Structure, Pumphouse</td></tr>
<tr><td>Containment</td><td>Drywell, Torus</td></tr>
<tr><td>Emergency Systems</td><td>NE, NW, SE Corner Rooms, HPCI Room, RCIC Room, RHR Valve Room, North CRD Area, South CRD Area, CSTs</td></tr>
<tr><td>Other</td><td>Control Building, Remote Shutdown Panel 1C388 Area, Panel 1C55/56 Area, SBGT Room</td></tr>
</table></td><td>Difference</td><td>Global Comment #8, 9, & 13
Same room/area listing as current EAL HU2</td><td>None</td></tr>
</table>

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**	Deviation	Global Comment #8, 9 & 13	V39
	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).	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the DAEC Security Shift Supervision.	Difference	Global Comment #9, 12, 14	None
	(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	Verbatim	Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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.	V39
	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

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.	V40

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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	V40
	(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.	V40

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	Intent and meaning of the EALs are not altered.	None

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 for 15 minutes or longer.	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 #
SU2	Recognition Category: SU2	SU3	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.	UNPLANNED loss of Control Room indications 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) 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.	(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.	Difference	Global Comment #12	None
	[BWR parameter list] / [PWR parameter list]: Reactor Power / Reactor Power RPV Water Level / RCS Level RPV Pressure / RCS Pressure Primary Containment Pressure / In-Core/Core Exit Temperature Suppression Pool Level / Levels in at least (site-specific number) two steam generators Suppression Pool Temperature / Steam Generator Auxiliary or Emergency Feed Water Flow	• Reactor Power • RPV Water Level • RPV Pressure • Primary Containment Pressure • Suppression Pool Level • Suppression Pool Temperature	Difference	Global Comment #4, 9 Intent and meaning of the EALs are not altered.	None

NEI parameter table (embedded in NEI 99-01 Rev. 6 column):

[BWR parameter list]	[PWR parameter list]
Reactor Power	Reactor Power
RPV Water Level	RCS Level
RPV Pressure	RCS Pressure
Primary Containment Pressure	In-Core/Core Exit Temperature
Suppression Pool Level	Levels in at least (site-specific number) two steam generators
Suppression Pool Temperature	Steam Generator Auxiliary or Emergency Feed Water Flow

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.	V41

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	V42
	(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	V42
	(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	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 #
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	V43
	(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 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
	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	V44
	(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	V44

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.	V44

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. AND	(1) a. AC power capability to 1A3 and 1A4 is reduced to a single power source for 15 minutes or longer. **AND**	Difference	Global Comment #9, 12	None
	b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.	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>
<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">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 of the following 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>• 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>
</table>

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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 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 • ECCS actuation • Thermal power oscillations greater than 10%	Difference	Global Comment #4, 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 #
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	V43
	(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 following 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	• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • River level above 757 feet • River Water Supply (RWS) pit low level alarm • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director	Difference	Global Comment #8 & 9	V45 V46

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.	V47
	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. • Loss of the safety function of a single train SAFETY SYSTEM.	Deviation	Adopted the revised EAL wording provided in approved EAL FAQ 2016-02; with the addition of a 3rd choice due to DAEC having single train SAFETY SYSTEM Intent and meaning of the EALs are not altered.	V47

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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 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	V43
	(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	V48 V49
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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.	V50 V51

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). **AND**	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4. **AND**	Difference	Global Comment #9 & 13	None
	b. EITHER of the following: • Restoration of at least one AC emergency bus in less than (site-specific hours) is not likely.	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)	• Reactor vessel water level cannot be restored and maintained above -25 inches.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V48

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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. **AND** b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer.	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 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 Difference	Global Comment #9, 12, 13 Global Comment #9 & 13 Intent and meaning of the EALs are not altered.	None V50 V51

APPENDIX A – ACRONYMS AND ABBREVIATIONS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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

DAEC DEVIATIONS AND DIFFERENCES MATRIX

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

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).	V22/47

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

LICENSE AMENDMENT REQUEST TSCR-166

SUPPORTING TECHNICAL INFORMATION

503 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.

licensee to promptly declare the emergency condition as soon as possible following the identification of the appropriate ECL. As used here, "promptly" means the next available opportunity unimpeded by activities not related to the emergency declaration, unless such activities are necessary for protecting health and safety. (See Paragraph 8 of this section.)

6. Consistent with the NRC's position that emergency declarations are made promptly, the final rule states that the 15-minute criterion not be construed as a grace period in which a licensee may attempt to restore plant conditions to avoid declaring an EAL that has already been exceeded. This statement does not preclude licensees from acting to correct or mitigate an off-normal condition, but once an EAL has been recognized as being exceeded, the emergency declaration shall be made promptly without waiting for the 15-minute period to elapse. This is particularly the case when the EAL threshold is exceeded based on occurrence of a condition, rather than the duration of a condition.

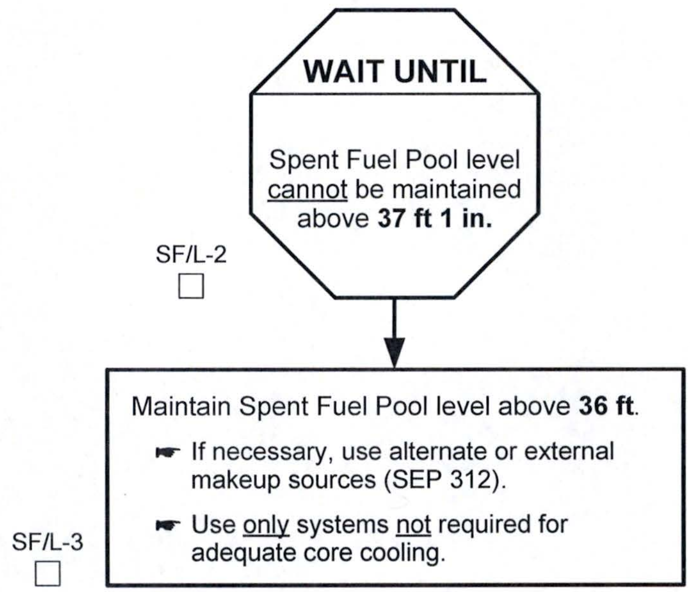
7. For EAL thresholds that specify a duration of the off-normal condition, the NRC expects that the emergency declaration process run concurrently with the specified threshold duration. Once the off-normal condition has existed for the duration specified in the EAL, no further effort on this declaration is necessary—the EAL has been exceeded. Consider as an example, the EAL "fire which is not extinguished within 15 minutes of detection." On receipt of a fire alarm, the plant fire brigade is dispatched to the scene to begin fire suppression efforts.

- If the fire brigade reports that the fire can be extinguished before the specified duration, the emergency declaration is placed on hold while firefighting activities continue. If the fire brigade is successful in extinguishing the fire within the specified duration from detection, no emergency declaration is warranted based on that EAL.

- If the fire is still burning after the specified duration has elapsed, the EAL is exceeded, no further assessment is necessary, and the emergency declaration would be made promptly. As used here, "promptly" means at the first available opportunity (e.g., if the Shift Manager is receiving an update from the fire brigade at the 15-minute mark, it is expected that the declaration will occur as the next action after the call ends).

- If, for example, the fire brigade notifies the shift supervision 5 minutes after detection that the brigade itself cannot extinguish the fire such that the EAL will be met imminently and cannot be avoided, the NRC would not consider it a violation of the licensee's emergency plan to declare the event before the EAL is met (e.g., the 15-minute duration has elapsed). While a prompt declaration would be beneficial to public health and safety and is encouraged, it is not required by regulation.

- In all of the above, the fire duration is measured from the time the alarm, indication, or report was first received by the plant operators. Validation or confirmation establishes that the fire started as early as the time of the alarm, indication, or report.



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.

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)

BASES

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)

BASES

BACKGROUND (continued)

connected loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for the DGs satisfy the intent of Safety Guide 9 as discussed in UFSAR Section 1.8.9 (Ref. 3). DGs 1G-31 and 1G-21 have the following ratings:

a. 2850 kW□continuous,

b. 3000 kW□2000 hours, and

c. 3250 kW□300 hours.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the UFSAR, Chapter 15 (Ref. 5), assume ESF Systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF Systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System and two separate and independent DGs

(continued)

BASES

LCO (continued)

(1G-31 and 1G-21) ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an Abnormal Operational Transient or a postulated DBA. Qualified offsite circuits are those that are described in the UFSAR, and are part of the licensing basis for the unit.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the essential buses. In accordance with commitments made in response to Generic Letter 2006-02 (Ref. 4), Condition C is entered whenever the grid operator (e.g., Midwest Independent System Operator (MISO)) determines that offsite power grid conditions are such that a trip of the DAEC turbine/generator would lead directly to voltages in the DAEC switchyard below the trip setpoints for Loss of Power (LOP) Instrumentation (LCO 3.3.8.1). The two offsite circuits consist of: 1) the incoming autotransformer (T1) and disconnect (1401, 6782, 2812 or 4731), the incoming circuit breaker (8490) and disconnect (8491), the underground 34.5 kV line, the standby transformer (1X4), the 4160 volt supply line and the two supply circuit breakers (1A301 and 1A401) to essential buses 1A3 and 1A4, respectively, and 2) either the incoming circuit breaker (5550) and disconnects (5551 and 5552) or incoming circuit breaker (5560) and disconnects (5553 and 5555), the overhead 161 kV line, the startup transformer (1X3), the two 4160 volt supply lines and the two supply circuit breakers (1A302 and 1A402) to essential buses 1A3 and 1A4, respectively.

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective essential bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the essential buses. Proper sequencing of loads, including non-essential load shedding capability, is a required function for DG OPERABILITY.

The AC sources must be separate and independent (to the extent possible) of other AC sources. For the DGs, the separation and independence are complete. For the offsite AC sources, the separation and independence are to the extent practical. A circuit may be connected to more than

(continued)

BASES

LCO (continued)

one essential bus, with slow transfer capability to the other circuit OPERABLE, and not violate separation criteria.

A circuit that is not connected to either essential bus is required to have OPERABLE slow transfer interlock mechanisms to both essential buses to support OPERABILITY of that circuit.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of Abnormal Operational Transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 4 and 5 and other Conditions in which AC sources are required are covered in LCO 3.8.2, "AC Sources — Shutdown."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the availability of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the

(continued)

BASES

ACTIONS

A.1 (continued)

second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

A.2

The power sources for the plant auxiliary power system are sufficient in number and have adequate electrical and physical independence to ensure that no single probable event could interrupt all auxiliary power at one time. In the condition of one inoperable offsite power source, all essential and non-essential buses remain OPERABLE and the remaining offsite power source continues to provide a highly reliable power source. Required Action A.2 requires restoring the inoperable offsite circuit to OPERABLE status, prior to entering MODE 2 from MODE 3 or 4. The inoperable offsite circuit must be restored to OPERABLE status prior to entering MODE 2 from MODE 3 or 4 to ensure that at least two offsite power sources will be available before the reactor is taken beyond just critical.

Entry into MODE 1 from MODE 2 with an inoperable offsite circuit is acceptable since LCO 3.0.4 allows continued operation of the unit in a MODE or other specified condition in which operation for an unlimited period of time is allowed. The inoperable offsite circuit only has to be repaired prior to entering MODE 2 from MODE 3 or 4.

B.1

To ensure a highly reliable power source remains with one DG inoperable, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions must then be entered.

(continued)

BASES

ACTIONS (continued)

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included). Failures of "redundant required features" refers to inoperable features associated with a division redundant to the division that has an inoperable DG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other division (Division 1 or 2) is inoperable.

If, at any time during the existence of this Condition (one DG inoperable), a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

(continued)

BASES

ACTIONS

B.2 (continued)

The remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

B.3

Required Action B.3 requires that the cause of the inoperability be evaluated to ensure a common cause failure does not exist that could render the OPERABLE DG inoperable. This evaluation may be performed by analysis or inspection or by demonstration of OPERABILITY. If the cause of inoperability exists on the other DG, it is declared inoperable upon discovery, and Condition D of LCO 3.8.1 is entered. Once the failure is repaired, and the common cause failure no longer exists, Required Action B.3 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG, SR 3.8.1.2 can be performed within the same Completion Time as Required Action B.3 to provide assurance of continued OPERABILITY of the remaining DG.

Conversely, Required Action B.3 may be satisfied by a simple review when the cause of the initial inoperability is pre-planned, preventive maintenance and testing. It is also permissible to perform elective maintenance as part of the pre-planned, preventive maintenance. In this case, there is no potential for a common cause failure, as no failure has occurred. At any point during the pre-planned maintenance or testing, if any new failure is detected, which on its own would cause the EDG to be inoperable and was not maintenance induced, the common cause evaluation must be re-performed on the Operable DG. If the 24 hour Completion Time for Required Action B.3 has expired at the point of discovery of the failure requiring corrective maintenance, Condition E must be entered until the common cause evaluation or SR 3.8.1.2 is performed.

(continued)

BASES

ACTIONS (continued)

B. 3 (continued)

In the event the inoperable DG is restored to OPERABLE status prior to completing B.3, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is a reasonable time to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG

B.4

To ensure the continued OPERABILITY of the remaining DG during the 7 day Completion Time of Required Action B.5, SR 3.8.1.2 must be performed once per 72 hours for the OPERABLE DG. The 72 hour Completion Time is acceptable since it has already been determined that a common cause failure does not exist.

Required Action B.4 is modified by a Note that removes this requirement when the cause of the initial inoperability is pre-planned, preventive maintenance and testing. It is also permissible to perform elective maintenance as part of the pre-planned preventive maintenance. In this case, no actual failure has occurred (i.e., a potential for a common mode failure has not been identified) and the likelihood of the other DG having an undetected failure during this period is low. At any point during the pre-planned maintenance, if any new failure requiring corrective maintenance is detected, i.e., which on its own would cause the EDG to be inoperable and was not maintenance induced, Required Action B.4 must be entered for the Operable DG. If the 72 hour Completion Time has expired at the point of discovery of the failure requiring corrective maintenance, then Corrective E must be entered until SR 3.8.1.2 is performed.

B.5

In Condition B, the remaining OPERABLE DG and offsite circuit(s) are adequate to supply electrical power to the onsite Class 1E Distribution System. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS

B.5 (continued)

The second Completion Time for Required Action B.5 establishes a limit based on the maximum time allowed for the combination of one DG and two offsite AC power sources to be inoperable during any single contiguous occurrence of failing to met the LCO except for Action A. If Condition B is entered while, for instance, two offsite circuits are inoperable and one circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 24 hours. This situation could lead to a total of 8 days, since initial failure of the LCO (except for Condition A), to restore the DG. At this time, the second offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 24 hours (for a total of 9 days) allowed prior to complete restoration of the LCO (except for Condition A). The 8 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions B and C are entered concurrently, and when corrective actions are completed prior to completing the shutdown required by LCO 3.0.3 (which is required to be entered by Action F). The "AND" connector between the 7 day and 8 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." This exception results in establishing the "time zero" at the time that the LCO was initially not met, instead of the time that Condition B was entered.

C.1 and C.2

As noted above, this Condition is entered whenever the grid operator informs the DAEC that a unit trip could lead to degraded voltage conditions in the DAEC Switchyard (Ref. 4).

Required Action C.1 addresses actions to be taken in the event of inoperability of redundant required features concurrent with inoperability of two offsite circuits. Required Action C.1 reduces the vulnerability to a loss of function. The Completion Time for taking these actions is 12 hours. When a concurrent redundant required feature failure exists, a shorter Completion Time of 12 hours is appropriate. These features are designed with redundant safety related divisions, (i.e., single division systems are not included in the list). Redundant required features failures consist of any of these features that are inoperable because any

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

inoperability is on a division redundant to a division with inoperable offsite circuits.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All offsite circuits are inoperable; and

b. A required feature is inoperable.

If, at any time during the existence of this Condition (two offsite circuits inoperable), a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

According to the recommendations contained in Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the Offsite Electrical Power System does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this degradation level:

a. The configuration of the redundant AC Electrical Power System that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

With both of the offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time in Required Action C.2 provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC Electrical Power System capable of meeting its design criteria.

According to the recommendations contained in Regulatory Guide 1.93 (Ref. 6), with the available offsite AC sources two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

The second Completion Time for Required Action C.2 establishes a limit based on the maximum time allowed for the combination of one DG and two offsite AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO except for Condition A. If Condition C is entered while, for instance, one DG is inoperable and the DG is subsequently restored OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total of 8 days, since initial failure of the LCO (except for Condition A), to restore one of the two inoperable offsite circuits. At this time, a DG could again become inoperable, one offsite circuit restored OPERABLE, and an additional 7 days (for a total of 15 days) allowed prior to complete restoration of the LCO (except for Condition A). The 8 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions B and C are entered concurrently, and when corrective actions are completed prior to completing the shutdown required by LCO 3.0.3 (which is required to be entered by Action F). The "AND" connector between the 24 hours and 8 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive must be met.

(continued)

BASES

ACTIONS (continued)

C. 1 and C.2 (continued)

This Completion Time allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." This exception results in establishing the "time zero" at the time that the LCO was initially not met (except for condition A), instead of the time that Condition C was entered.

D.1

With two DGs inoperable, there is no remaining standby AC source. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for the majority of ESF equipment at this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown. (The immediate shutdown could cause grid instability, which could result in a total loss of AC power.) Since any inadvertent unit generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation. According to the recommendations contained in Regulatory Guide 1.93 (Ref. 6), with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

E.1 and E.2

If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the associated Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

F.1

Condition F corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC Electrical Power System will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with UFSAR Section 3.1.2.2.9 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are largely in accordance with the recommendations of Safety Guide 9 as discussed in UFSAR Section 1.8.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10) or as addressed in the UFSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following summary is generally applicable. The minimum steady state output voltage of 3744 V is approximately 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1 (Ref. 11), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. This value also provides a large margin of safety, since safety related motors are capable of accelerating their loads at 70% of rated voltage (2912 V or 322 V). The specified maximum steady state output voltage of 4576 V or 110% of 4160 V is less than the maximum operating voltage specified for 4000 V motors (4756 V), and therefore also provides ample margin. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 59.5 Hz and 60.5 Hz, respectively. These values are approximately equal to ± 1% of the 60 Hz nominal frequency and are conservative with respect to the recommendations found in Regulatory Guide 1.9 (Ref. 17).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that at least the minimum required offsite power supply breakers are in their correct position to ensure that distribution buses and loads are connected to either the preferred power source or the alternate preferred power source and that appropriate independence of offsite circuits is maintained. This can be accomplished by verifying that an essential bus is energized, and that the status of offsite supply breakers that are displayed in the control room are correct. The status of manual disconnects is verified administratively. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs have been modified by a Note (Note 2 for SR 3.8.1.2 and Note 1 for SR 3.8.1.7) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and (for SR 3.8.1.2 only) followed by a warmup prior to loading. Note 3 to SR 3.8.1.2 allows delaying the entry into associated Conditions and Required Actions for up to two hours during the performance of the conditional surveillance required by Required Actions B.3 or B.4. This Note is necessary because to perform a slow start and warmup of the DG requires reducing the governor control setting to minimum and securing the generator field excitation. The governor control setting is gradually increased to bring the DG to synchronous speed and to allow for warmup. Once the DG is at synchronous speed, the generator field excitation is enabled and the DG is again capable of supplying the essential bus. During this warmup portion of the surveillance test, the DG is incapable of supplying the essential bus and is considered inoperable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

After completion of the SR, the fuel racks to the DG are disabled to allow purging of any residual fuel oil from the cylinders. This also renders the DG inoperable. The two hours allowed by the Note minimizes the amount of time a DG is inoperable while providing enough time to perform the required Conditional Surveillance and avoids entering the shutdown actions of Condition E or F unnecessarily.

For the purposes of this testing, the DGs are manually started from standby conditions. Standby conditions for a DG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines during testing, the manufacturer of the DGs installed at the DAEC recommends a modified start in which the starting speed of the DG is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 2 (SR 3.8.1.2).

SR 3.8.1.7 requires that DG starts from standby conditions and achieves required voltage and frequency (i.e. - voltage ≥ 3744 V and frequency ≥ 59.5 Hz) within 10 seconds; and achieves steady state voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 59.5 Hz and ≤ 60.5 Hz. The 10 second start requirement supports the assumptions in the design basis LOCA analysis of UFSAR, Section 15.2.1 (Ref. 12). The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 3 of SR 3.8.1.2), when a modified start procedure as described above is used. If a modified start is not used, the 10 second start requirement of SR 3.8.1.7 applies. In addition to the SR requirements, the time for the DG to reach steady state operation, unless the modified DG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The normal Frequency for SR 3.8.1.2 is consistent with Safety Guide 9. The Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing and can be manually loaded to ≥ 2750 kW and ≤ 2950 kW, providing a 200 kW range centered on the continuous duty rating of the DGs of 2850 kW. This range ensures that the DGs are tested at a load above the maximum expected accident load. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor greater than 0.9 lagging. While a value of 0.8 is the design rating of the machine, the machine is operated at power factors greater than 0.9 for normal operations and greater than 0.8 for surveillance testing. The load limit is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The normal Frequency for this Surveillance is consistent with Safety Guide 9.

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test.

Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which the day tank low level alarm is annunciated. This low level alarm should only be received if the automatic fuel oil transfer instrumentation is not functioning properly. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of approximately one hour of DG operation at full load, considering a conservative fuel consumption rate. Verification that at least a one hour supply of fuel oil exists in a day tank provides assurance that a DG can operate continuously, and also allows the operating crew sufficient time to take corrective action should the automatic fuel oil transfer system not function properly.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Testing for water content and removal of water from the fuel oil day tanks as necessary, eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water, as necessary, minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequencies meet the intent of Regulatory Guide 1.137 (Ref. 10). This SR is for preventive maintenance. The presence of water does not necessarily represent a failure of this SR provided that accumulated water is removed during performance of this Surveillance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. It is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for manual fuel transfer systems are OPERABLE. Additional assurance of fuel oil transfer pump OPERABILITY is provided by meeting the testing requirements for pumps that are contained in the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 13). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

The slow transfer of each 4.16 kV essential bus power supply from the preferred offsite circuit (i.e. - the startup transformer) to the alternate preferred offsite circuit (i.e. the standby transformer) demonstrates the OPERABILITY of the alternate preferred circuit distribution network to power the shutdown loads. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of the Surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed on this Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the Electrical Distribution Systems that could challenge continued steady state operation and, as a result, plant safety systems. Credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and the capability to reject the largest single load and return to the required voltage and frequency (i.e. - voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 59.5 Hz and ≤ 60.5 Hz) within predetermined periods of time (i.e., 1.3 seconds for voltage and 3.9 seconds for frequency) while maintaining an acceptable margin to the overspeed trip. The largest single load for each DG is a core spray pump motor (700 hp). This Surveillance may be accomplished by tripping its associated single largest post-accident load with the DG solely supplying the bus.

As specified by IEEE-308 (Ref. 14), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower. For both DGs, this represents 64.5 Hz, equivalent to 75% of the difference between nominal speed and the overspeed trip setpoint.

The time, voltage, and frequency tolerances specified in the Bases for this SR are derived from UFSAR Table 8.3-1 (Ref. 16) recommendations for response during load sequence intervals. The voltage and frequency are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are the steady state voltage and frequency to which the system must recover following load rejection within a predetermined time period. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9).

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the Electrical Distribution Systems that could challenge continued steady state operation and, as a result, plant safety systems. Credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.10

This Surveillance demonstrates that DG non-critical protective functions (e.g., high jacket water temperature and low lubricating oil pressure) are bypassed on either an ECCS initiation test signal or a LOOP test signal and critical protective functions (engine overspeed and generator differential current) trip the DG to avert substantial damage to the DG unit. The non-critical trips are bypassed during DBAs and LOOPs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on engineering judgment, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at this Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.11

As specified by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and load transfer from the DG to the offsite source can be made and that the DG can be returned to ready-to-load status when offsite power is restored. The DG is considered to be in ready-to-load status when the DG is at rated speed and voltage, the output breaker is open and can receive an auto-close signal on bus undervoltage, and the individual pump timers are reset.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), and takes into consideration plant conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.12

Under either LOCA conditions or during a loss of offsite power, loads are sequentially connected to the bus by a timed logic sequence using individual time delay relays. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. Verifying the load sequence time interval is greater than or equal to 2 seconds ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load. The Allowable Values for the Core Spray and Low Pressure Coolant Injection Pump Start - Time Delay Relays, Table 3.3.5.1-1, Functions 1.e and 2.e, ensure this time interval is maintained as well as ensuring that safety analysis assumptions regarding ESF equipment time delays are not violated. Allowances for instrument inaccuracies in the load sequence time interval are also accounted for by the Pump Start - Time Delay Relay Allowable Value.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2); takes into consideration plant conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.13

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF Systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates DG operation during a Loss of Offsite Power actuation test signal (LOOP signal) in conjunction with an ECCS initiation signal (LOCA signal). This test verifies all actions encountered from the LOOP/LOCA, including the LOOP/LOCA load shedding function and energization of the essential buses and respective loads from the DG. This Surveillance also demonstrates the as-designed operation of the standby power sources during a LOOP, including: 1) de-energization of the essential buses, 2) the dead bus load shedding function, and 3) that the DG receives a start signal. This surveillance also demonstrates that the DG automatically starts from the design basis actuation signal (LOCA signal). It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency (i.e., voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 59.5 Hz and ≤ 60.5 Hz) within the specified time (10 seconds). In lieu of multiple demonstrations of DG starting and achieving the required voltage and frequency in the specified time from each of the various start signals (LOOP, LOCA and LOOP/LOCA), and operation for ≥ 5 minutes, testing that adequately shows the capability of the DG to start from each of the signals is acceptable. The DG auto-start time of 10 seconds is derived from requirements of the accident analysis, (Ref. 12), for responding to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes (with a LOOP signal in conjunction with a LOCA signal present) in order to demonstrate that all of the starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.13 (continued)

loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, or systems are not capable of being operated at full flow. In lieu of actual demonstration of connection and loading of these loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that proper operation with each of the various signals present is verified.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length. Operating experience has shown that these components usually pass the SR when performed at this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove the required offsite circuit from service, perturb the Electrical Distribution System, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. UFSAR, Section 3.1.2.2.8.
2. UFSAR, Section 8.2.1.3 and Section 8.3.1.1.5
3. UFSAR, Section 1.8.9.
4. FPL letter, L-2006-073, dated April 3, 2006, Response to NRC Generic Letter 2006-02, Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power.
5. UFSAR, Chapter 15.

(continued)

BASES

REFERENCES (continued)

6. Regulatory Guide 1.93.

7. Generic Letter 84-15.

8. UFSAR, Section 3.1.2.2.9

9. Regulatory Guide 1.108.

10. Regulatory Guide 1.137.

11. [Deleted]

12. UFSAR, Section 15.2.1

13. ASME Boiler and Pressure Vessel Code, Section XI.

14. IEEE Standard 308.

15. [Deleted]

16. UFSAR, Table 8.3-1.

17. Regulatory Guide 1.9.

FOLLOW-UP ACTIONS (continued)

18. Direct an operator and electricians (if available) to perform Attachment 10 Alternate AC power to 125VDC and 250VDC chargers. ________

NOTE

The inverters will automatically trip at 105 VDC decreasing.

____

19. After one hour has elapsed, dispatch an operator to implement Attachment 13, Load Shedding to Preserve Station Batteries (from AOP 301.1 hanging file). Attachment 13 is to be completed within two hours of the SBO event. ________

20. **IF** a SBDG is available for operation, **THEN** restore power to its essential bus per Restoration of Standby Diesel Generator Power section. ________

21. **WHEN** the DAEC Switchyard inspection has been completed, **THEN** restore power to the switchyard per Restoration of Offsite Power section. ________

22. **WHEN** sufficient offsite power becomes available, **THEN** restore power to the non-essential buses per AOP 304.1. ________

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.

FIGURE F-5
DAEC TELEPHONE SYSTEMS

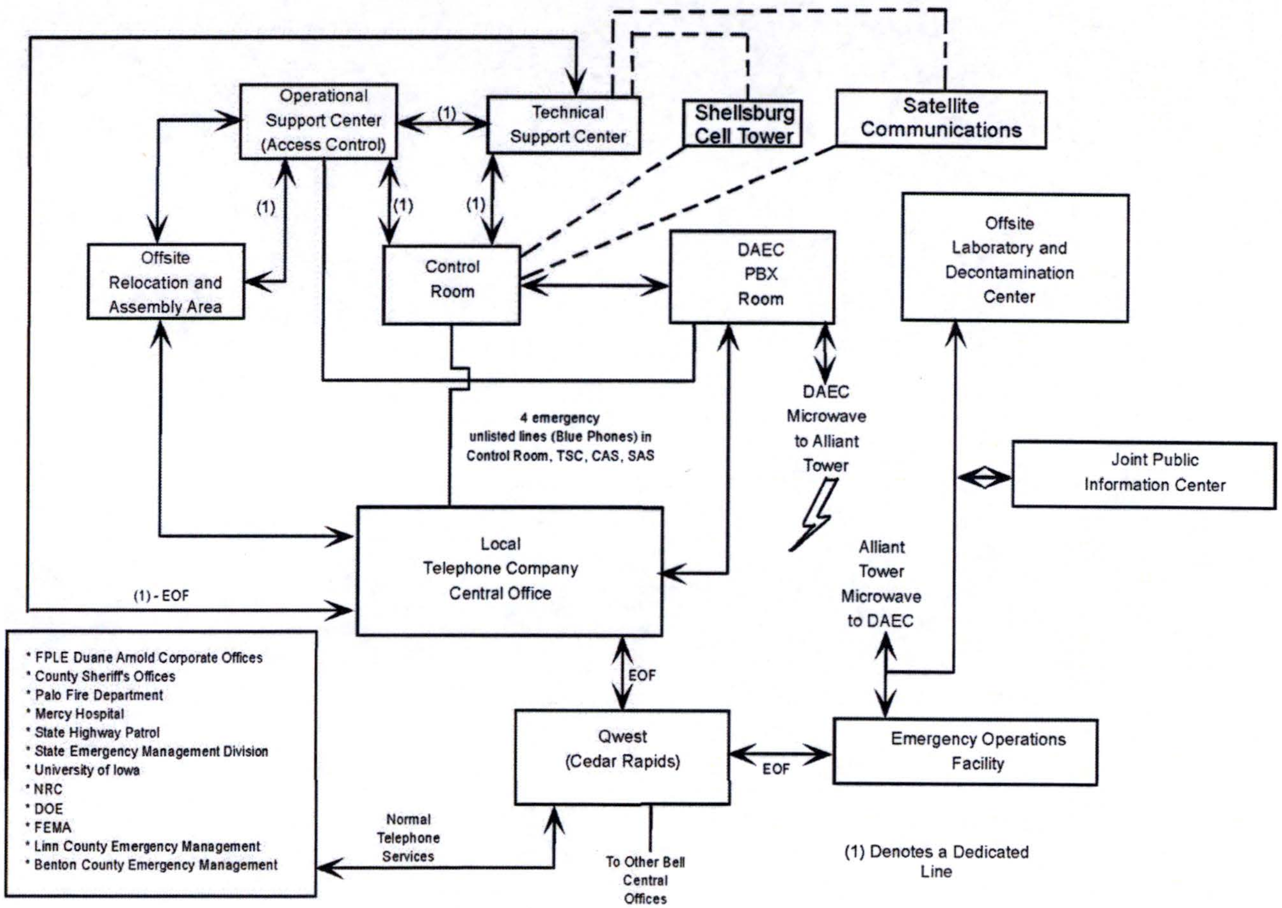


FIGURE F-6
FEDERAL TELEPHONE SYSTEM (FTS-2001)

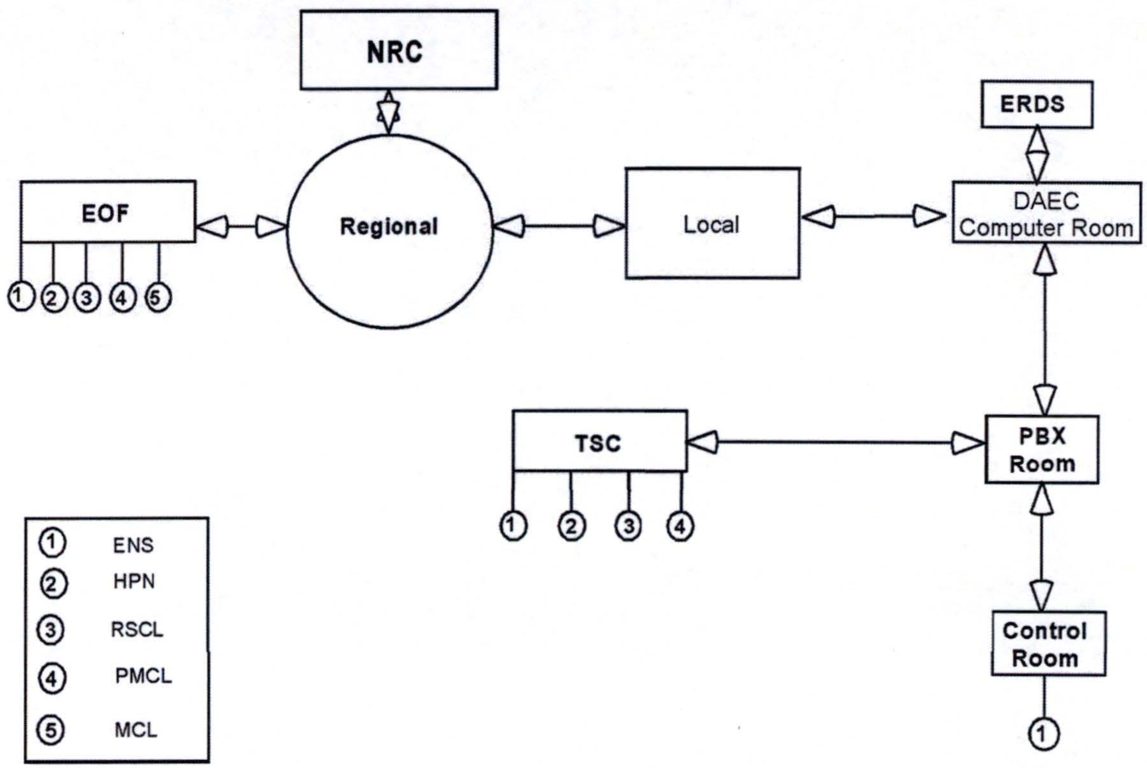
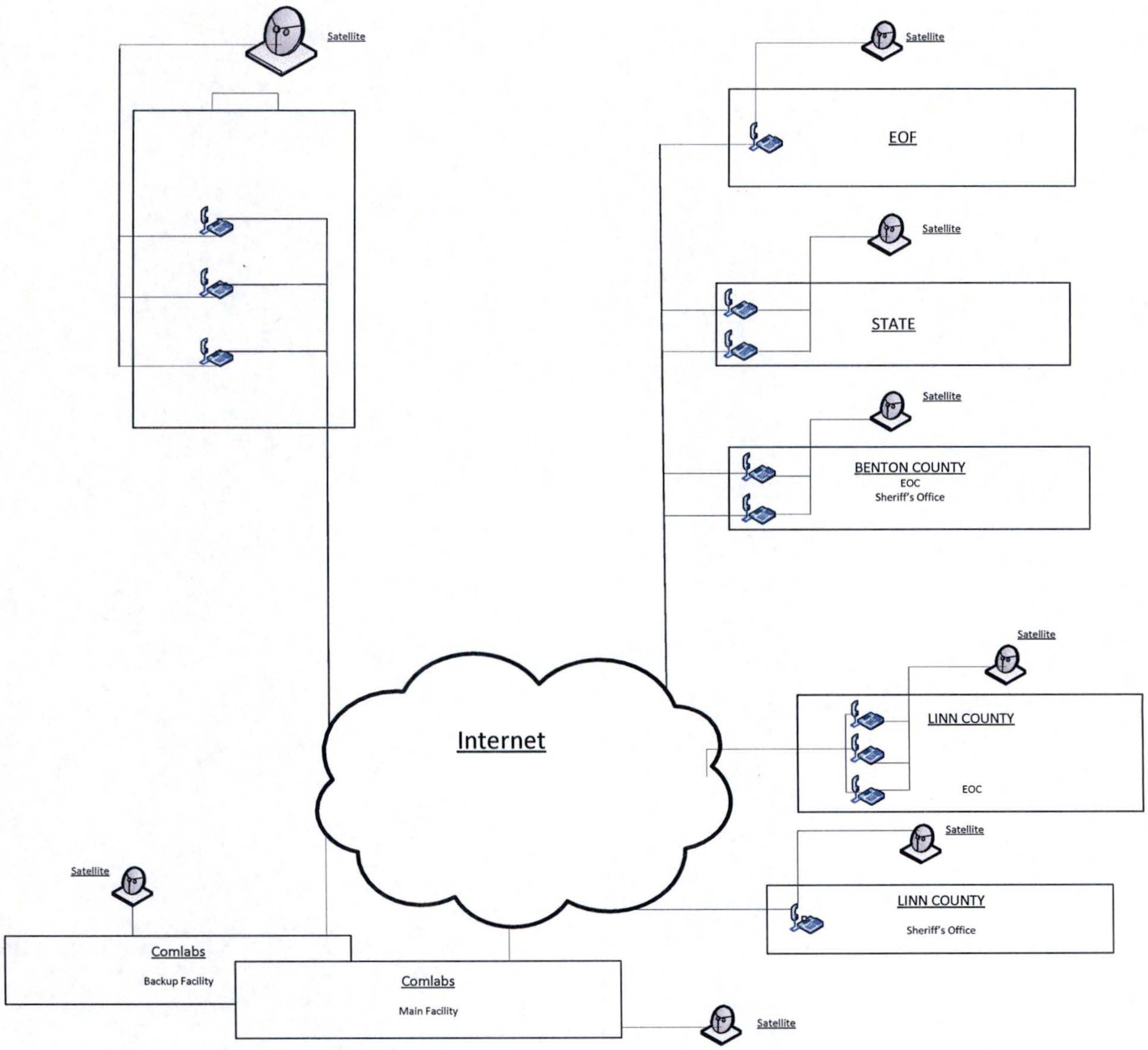


FIGURE F-7
ALL-CALL TELEPHONE SYSTEM



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.

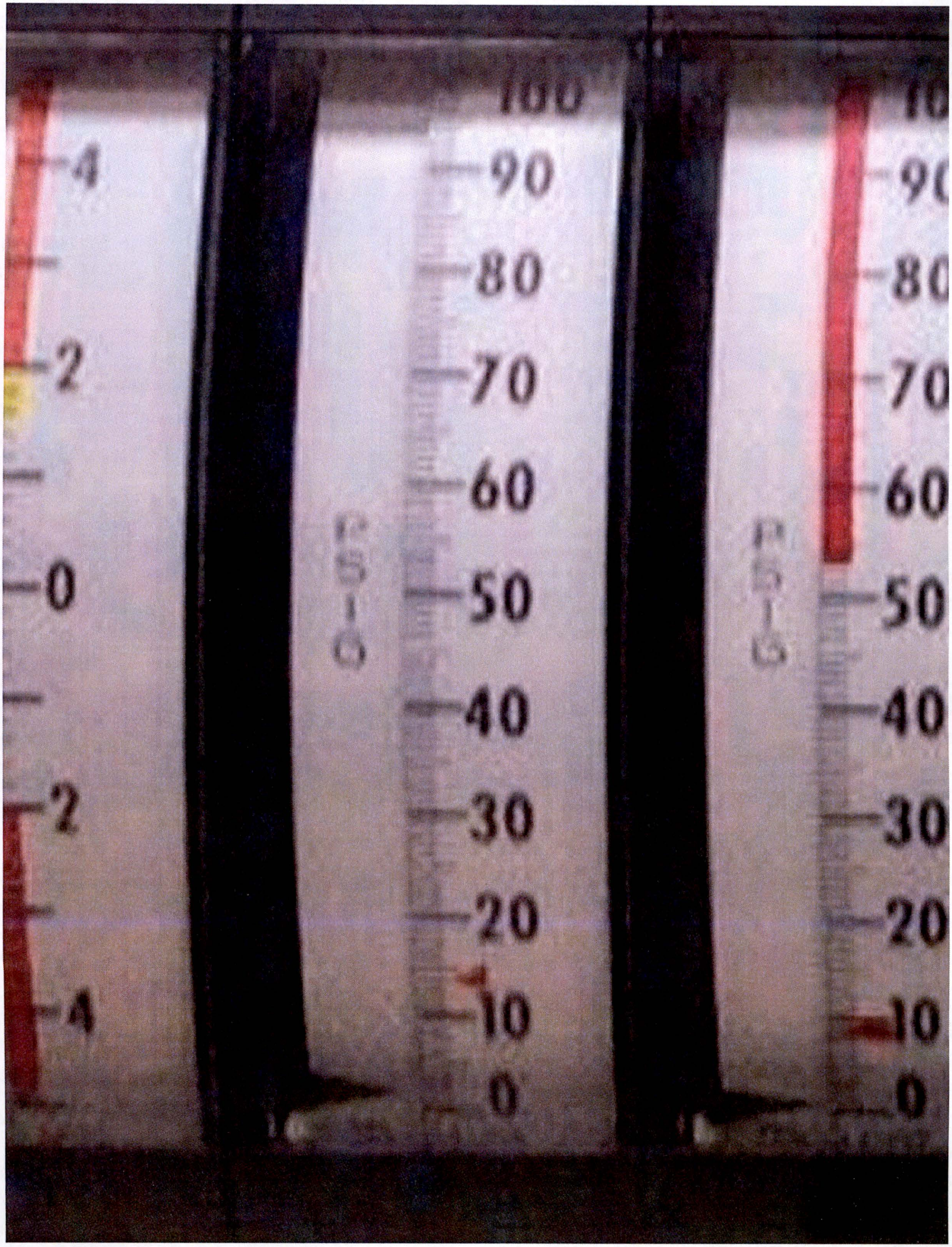
(b) Local and state agencies notified by commercial communication system (telephone or facsimile) may require verification of the identity and authenticity of the caller and the message received.

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.


4
2
0
2
4
100
90
80
70
60
PSIG
50
40
30
20
10
0
PSIG

ACUREN

Acuren USA
BPX GPBU
Prudhoe Bay, Alaska
99734-0122
Phone: (907) 659-8249
Fax: (907) 659-5936
Materials Engineering and Testing
A Rockwood Company

November 17, 2012

RE: Radiation Incident, License No. **42-32443-01**
Reported to NRC by Telephone per 10 CFR, §30.50(b)(2)(ii), 1050hrs, November 17,2012, Report #48516

Equipment Problem:
Source could not be fully retracted into shielded position.

Cause of incident:
The control cable of a 35' control assembly broke approximately three (3) inches from the source connector.

Equipment Involved:
7' flexible guide tube serial number GT1211
Exposure device model: Sentinel Delta 880D, serial # D3503
Isotope : Ir-192, source serial # 87400B
Source Activity : 60.7ci
35' Control Assembly s/n 11645

Place, Date, Time:
Place: Drill Site 6 to Flow Station 3 Access Road, Eastern Operating Area, Prudhoe Bay, Alaska
Date: November 16, 2012
Time: 1400 hrs AKT

Actions taken to establish normal operations:
The source assembly was gravity fed from the guide tube onto the ground and then shielded by reverse placement into an 880D exposure device, s/n D3926. A serviceable control assembly was then connected to exposure device D3503 and control cable routed through the device and guide tube, then connected to the source. The source was then retracted into device D3503 without incident.

Corrective actions taken and planned to prevent reoccurrence:
Remove 100% of control assemblies from service and complete a thorough inspection. Perform a safety stand down with all radiographers and assistants for review of daily equipment inspections as required in the Acuren O&E manual.

Qualifications of personnel involved in incident:
(1) Texas Industrial Radiographer Certification holder, dose received, 20mR
(2) IRRSP card holder, dose received, 20mR
(3) IRRSP card holder, Source Retrieval qualified, dose received, 48mR
(4) IRRSP card holder, Source Retrieval qualified, dose received, 95mR

Robert L. Jefferson
Radiation Safety Officer, Alaska
Acuren USA Inc.
PO Box 340122
Prudhoe Bay, AK 99734
direct: 907-659-8249

ABNORMAL OPERATING PROCEDURE

AOP 410

LOSS OF RIVER WATER SUPPLY/HIGH RIVER BED ELEVATION/LOW RIVER WATER DEPTH

Usage Level
REFERENCE

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:

NOTE

Refer to EPIP 1.1 for EAL ASSESSMENT.

NOTE

The portions of this procedure that minimize river make-up flow and maximizes make-up flow from other sources may be used as necessary in the event of an intrusion of excess foreign material from the Cedar River and/or Intake Structure without a Loss of River Water Supply.

IMMEDIATE ACTIONS

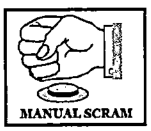
1. None

AUTOMATIC ACTIONS

- CV-4914 and CV-4915 open and CV-4910A and CV-4910B close on low level in the ESW/RHRSW wet pits

FOLLOW-UP ACTIONS

1. Establish critical parameter monitoring of Circ Pit Level and ESW/RHRSW Pit Level, as priorities allow. ________

2.	**IF**	power is available **AND** annunciator 1C06A (A-1[2]) "A"["B"] RWS PIT LO LEVEL is not on	**THEN**	attempt to start standby pumps in both RWS Subsystems as necessary to restore needed makeup flow.	________
3.	**IF**	standby pumps did not start	**THEN**	attempt to restart the tripped River Water Pumps.	________
4.	**IF**	power is not available	**THEN**	attempt to restore power from 1C08 **AND** attempt to restart pumps.	________
5.	**IF**	no RWS pumps can be started 	**THEN**	Reduce recirc flow to 39 Mlbm/hr in accordance with IPOI 4 **AND** manually scram the reactor.	________
6.	**IF**	offsite power was lost **AND** River Water Supply pumps fail to start from the diesel	**THEN**	Attempt to start pumps manually, that have their start permissive light on. **AND** restore RWS makeup to the stilling basin.	________

7. Update the Online Risk Monitor for the status of River Water Supply Pumps. ________
8. Maximize Well Water flow for makeup to the Circ Pit, while maintaining ≤ 170 psig well water system pressure at the main plant BEECO Backflow Preventer. ________
9. Close the Circ Water Inlet to Blowdown Line valve MO-4253 to maintain Circ Water Pit inventory. ________
10. Secure Cooling Tower Fans as allowable ________
11. Secure one circ water pump and one cooling tower as soon as possible. ________

FOLLOW-UP ACTIONS (continued)

12. Send an operator to the Intake Structure to verify alarms and check RWS pump breaker condition. ________

NOTE

The RHRSW Pumps should be kept in operation to support use of the RHR System as directed by EOPs (Torus Cooling/Shutdown Cooling). The RHRSW Pumps may be secured when it is determined that ESW/RHRSW pit level cannot be maintained above 4 feet. Securing RHRSW Pumps prior to reaching 4 feet in the pits will preserve the ESW Supply to the SBDGs. ________

13. Minimize use of water from the RHRSW/ESW pit as follows:

    a. Secure RHRSW pumps unless required to support operation of the RHR System ________

    b. Shutdown any SBDG not required to ensure one Essential Bus is energized and/or required to ensure adequate core cooling. ________

        (1) Verify SBDG Cooling Valves CV-2080 and CV-2081 close when the respective SBDG is secured. ________

    c. Verify Well Water is available for cooling the operating Control Building Chiller and then secure ESW to the Control Building Chillers by unlocking and closing V-13-122 and V-13-125 on the Reactor Building 812' level. ________

14. Minimize heat addition to the Torus (Reliefs, HPCI, RCIC). ________

15. Use the Turbine Bypass valves and/or Bypass Jack for Reactor pressure control and Reactor cooldown and continue to bleed steam to the Main Condenser for as long as possible. ________

16. At 1C15 and 1C17, place the HI COND BACKPRESS BYPASS switches in BYPASS. ________

17. Notify Security at 7254 prior to opening Pumphouse doors to arrange for required Security compensatory measures. ________

FOLLOW-UP ACTIONS (continued)

18. Establish makeup to the ESW/RHRSW Wet Pits from the Fire System using 2-1/2" and/or 5" hoses as follows: **{C001}** ________

    a. Using 2-1/2" hoses (N/A if not used):

        (1) Notify Mechanical Maintenance to remove the cover from outside hose head (octopus head). ________

        (2) Obtain 2-1/2" hoses from the warehouse and fire brigade trailer, and rig as many as possible (8 preferred) from the octopus head to the stilling basin. ________

        (3) Lash the hoses together with rope to prevent hose whip when the lines are charged. ________

    b. Using 5" hoses (N/A if not used):

        (1) Obtain 5" hoses from the B5b hose trailer. ________

        (2) Connect a 5" hose to any of the following fire hydrants as required: ________

        - FH-1 located east of the Turbine Building
        - FH-2 located southeast of the Turbine Building
        - FH-7 located northeast of the Turbine Building

        (3) Rig 5" hoses as needed to the Stilling Basin. ________

        (4) Lash the hoses together with rope to prevent hose whip when the lines are charged. ________

    c. When directed by the CRS, start 1P-48 or 1P-49 and valve in hoses as necessary to maintain RHRSW/ESW pit level. ________

    d. Monitor RHRSW/ESW pit level at 1C29, computer points B279 and B280, or on group display AOP 410. ________

FOLLOW-UP ACTIONS (continued)

19. Establish makeup to the RHRSW/ESW pits from GSW as follows:

    a. Verify at least one GSW pump is running. ________

    b. At 1C452 (located in the B RHRSW/ESW pump room) open the following valves with the appropriate handswitch: ________

CV-8035A	A RHRSW/ESW WET PIT CHLORINE INJECTION ISOLATION	HS-8035A
CV-8035B	B RHRSW/ESW WET PIT CHLORINE INJECTION ISOLATION	HS-8035B
CV-8034	RHRSW/ESW DILUTION WATER SUPPLY VALVE	HS-8034

    c. In the CHLORINE BOOSTER PUMP ROOM, note and record the position of V-80-154, then fully open V-80-154 GSW Dilution Water Supply Balancing valve using a wrench from the tool board. ________

NOTE

NPSH requirement is 8 feet for the Circ Water Pumps. NPSH requirement is 4 feet for the ESW, RHRSW, and GSW pumps. ________

20. Monitor the Circ Water Pit level at Computer Point F092 and secure Circ Water Pumps if level cannot be maintained or restored greater than 8 feet and GSW pumps as necessary to prevent cavitation. ________

21. Monitor RHRSW/ESW Pit level and secure RHRSW and ESW pumps as necessary to prevent cavitation. ________

22. When the status of each River Water Supply pump is known and there is at least one operable pump in each RWS loop, select the operable pumps on HSS-2911A and B to be the pumps to auto restart on the diesel. ________

23. Comply with Technical Specifications for River Water Supply. ________

24. When River Water Supply pump operation is restored, return system to normal operation per OI 410. ________

25. Update the Online Risk Monitor for the status of River Water Supply Pumps. ________

FOLLOW-UP ACTIONS (continued)

26. Open and Lock open the following valves:
    - V-13-122 ESW Loop A Return Header Isolation ______
    - V-13-125 ESW Loop B Return Header Isolation ______
27. Independently verify the following valves are locked open:
    - V-13-122 ESW Loop A Return Header Isolation ______ IV
    - V-13-125 ESW Loop B Return Header Isolation ______ IV

PROBABLE ANNUNCIATORS

1C06A,	A-1	"A" RWS PIT LO LEVEL
	A-2	"B" RWS PIT LO LEVEL
	A-3	"B" RWS PUMP 1P-117B TRIP
	A-4	"D" RWS PUMP 1P-117D TRIP
	B-1	"A" RWS PUMP 1P-117A TRIP
	B-2	"C" RWS PUMP 1P-117C TRIP
	B-5	"A" COOLING TOWER BASIN HI/LO LEVEL
	B-6	"B" COOLING TOWER BASIN HI/LO LEVEL
	D-1	"A" RHRSW/ESW PIT LO LEVEL
	D-2	"B" RHRSW/ESW PIT LO LEVEL
	D-11	CIRC WATER PIT LO LEVEL

PROBABLE INDICATIONS

<u>1C06</u>

– River Water makeup flow stopped at FR-4916 and FR-4917 or FI-4916 and FI-4917

HIGH RIVER BED ELEVATION

IMMEDIATE ACTIONS

1. None

AUTOMATIC ACTIONS

- None

FOLLOW-UP ACTIONS

NOTE

River water elevation is measured via LR-2901, Computer Point M010.V, and/or STP 3.0.0-01. River water depth is determined utilizing 1H239 River Depth Measuring Crane per OI 410 or during surveillance testing. River bed elevation is determined by subtracting the 1H239 measured river water depth from river water elevation. River bedrock elevation ranges from 722' to 723'. The difference between river bed elevation and river bedrock elevation is considered the sand bed height. River bed elevation and sand bed elevation are synonymous with each other.

Current sand gate elevation is listed in STP 3.0.0-01.

Actions taken in this section are to be coordinated with the requirements of the Low River Water Depth section of this AOP.

1. **IF** River bed elevation is determined to be greater than (>) 726' utilizing 1H239. **THEN** Coordinate with the Work Week Manager, System Engineering, and Maintenance to perform the following as applicable.

   a. Task and schedule Iowa Vane Field Mapping (Model WO 40105684).

   b. **IF** following completion of Iowa Vane Field Mapping it is determined river bed elevation is less than (<) 727' within 10' of the intake, **THEN** continue performing Iowa Vane Field Mapping every two weeks.

   c. **IF** following completion of Iowa Vane Field Mapping it is determined river bed elevation is greater than or equal to (≥) 727' within 10' of the intake, **THEN** perform Iowa Vane Field Mapping weekly.

   d. **IF** following completion of Iowa Vane Field Mapping it is determined river bed elevation is greater than or equal to (≥) 728' within 10' of the intake,

   OR

   IF River bed elevation is determined to be greater than (>) 727' utilizing 1H239,

   THEN perform the following:

   (1) Determine river water depth and river bed elevation daily utilizing 1H239 per OI 410, River Water Depth Section. Log daily river water depth and river bed elevation results in the shift log.

   (2) Task and schedule River Channel Mapping (Model WO 1374301).

   (3) OSM and System Engineering shall evaluate raising the sand gate elevation to minimize sand ingestion based on current conditions and trend. Sand gates may be raised per Step 3.

FOLLOW-UP ACTIONS (continued)

2. **IF** River bed elevation is determined utilizing 1H239 to be less than or equal to (≤) 0.5 feet from the current sand gate elevation. **THEN** work with the Work Week Manager, System Engineer, and Maintenance to perform the following as applicable. ______

   a. **IF** River Bed Elevation is less than (<) 0.5 feet below the top of the Sand Gate, **THEN** update the Online Risk Monitor by using the Review/Change Environmental Variables button to adjust Ultimate heat Sink to High Risk. ______

   b. Determine river water depth and river bed elevation daily utilizing 1H239 per OI 410, River Water Depth Section. Log daily river water depth and river bed elevation results in the shift log. ______

   c. Task and schedule the following:

      (1) Iowa Vane Excavation (Model WO 40158555): ______

      - Schedule excavation to begin prior to river bed elevation reaching the top elevation of the sand gates.
      - Excavate Iowa Vane field and river channel downstream of intake structure for a minimum of 100 yards.
      - Excavate Iowa Vane field to river bottom (722' to 723') per mapping.
      - Monitor screen differential pressure during excavation.

NOTE ______

Dive Work Orders are to be performed following completion of excavation. Inspection/cleaning includes Forebay up to the river side of Lite Locs. Removal of Lite Locs and inspection/cleaning to Traveling Screen and RWS pump pits to be completed based on Forebay results and OSM direction.

      (2) Diving inspection and cleaning work orders: ______

      - 'A' side Intake structure forebay (Model WO 1374681) or 'B' side Intake structure forebay (Model WO 1375828) as applicable.
      - 'A' side RWS pump pit (Model WO 1374679) and 'B' side RWS pump pit (Model WO 1374680) per OSM direction.

      (3) Following excavation and diving, determine river water depth and river bed elevation using 1H239 per OI 410 River Water Depth section. Log river water depth and river bed elevation results in the shift log. ______

      (4) Following excavation and diving, return to Step(s) 1 and/or 2 and take actions as applicable. ______

FOLLOW-UP ACTIONS (continued)

NOTE

Base sand gate elevation is 728.0'. Fully raised sand gate elevation is 730'. Fully lowered elevation is 723.5'. Both sand gates are to be maintained at the same elevation.

Raising sand gates will impact water level at the intake structure.

Divers must inspect and clean (if necessary) the intake structure forebay in the vicinity of the Sand Gates to verify the gates are clear prior to lowering the gates back to 'base' elevation.

Sand Gate electrical controls are non-functional. Sand Gates must be raised and lowered manually using manual ratchet tools.

Sand loading on the river side of the Sand Gates may impede or prevent gate movement.

The OSM and System Engineering shall consider the requirements of SR 3.7.2.3 and SR 3.7.2.5 for maintaining greater than 1' water level above the sand gate elevation when proposing to raise the sand gates.

3. **IF** desired to raise sand gate elevation, **THEN** coordinate with the Work Week Manager, System Engineering, and Maintenance to perform the following as applicable.

   a. OSM consult with System Engineering to determine the newly proposed sand gate elevation. ______

      (1) **IF** it is expected that river water level will remain greater than or equal (≥) to 2.5' above the newly proposed sand gate elevation, **THEN** the sand gates may be raised. ______

      (2) **IF** it is expected that river water level will not remain greater than or equal (≥) to 2.5' above the newly proposed sand gate elevation, **THEN** the sand gates may be raised only after the OSM verifies that the requirements of SR 3.7.2.3 and SR 3.7.2.5 will remain met. ______

      (3) **IF** sand gates are to be raised, **THEN** perform the remaining steps, otherwise N/A the remaining Step 3 substeps. ______

FOLLOW-UP ACTIONS (continued)

NOTE

The area beneath the sand gates should be inspected and cleaned as necessary to remove sand accumulation that may prevent sand gate movement prior to attempting to lower the sand gates.

CAUTION

If sand gates are raised, sand may build up underneath the gates preventing the gates from being lowered. This could result in a loss of river water supply if river level drops below the top of the gates. To preclude this, the sand gates should be kept at least 1 foot below river water level as monitored in STP 3.0.0-01. **{C002}**

b. **IF** not already completed, **THEN** task and schedule Model WO 1374681 **OR** Model WO 1375828 for diver inspection/cleaning of the Intake Structure forebay. ________

c. Manually raise gates to the proposed elevation per OI 410 (maximum elevation is 730.0'). ________

d. Verify final sand gate elevation using gate dial indicator, gate height measuring device, or divers. ________

e. Determine river water depth and river bed elevation utilizing 1H239 per OI 410, River Water Depth Section. Log river water depth and river bed elevation results in the shift log. ________

f. Revise STP 3.0.0-01, Instrument Checks, to document final current sand gate elevation. ________

g. Continue to perform applicable actions per Step(s) 1 and/or 2 until the conditions of 3.i are met. ________

h. **WHEN** River Bed Elevation is more than (>) 0.5 feet below the top of the Sand Gate, **THEN** update the Online Risk Monitor by using the Review/Change Environmental Variables button to adjust Ultimate heat Sink to Normal Risk. ________

FOLLOW-UP ACTIONS (continued)

i. **IF** river bed elevation is determined to be less than (<) 726' utilizing 1H239.

AND

IF river bed elevation is determined utilizing 1H239 to be greater than (>) 0.5' below current sand gate elevation.

THEN the sand gates may be returned to 728' by performing the following

(1) Verify all excavation, inspection and or cleaning as required by Step 2 and or Step 3.b is completed as necessary. ________

(2) Manually lower sand gates to base elevation of 728' or as determined by System Engineering per OI 410. ________

(3) Verify final sand gate elevation using gate dial indicator, gate height measuring device, or divers. ________

(4) Determine river water depth and river bed elevation utilizing 1H239 per OI 410, River Water Depth Section. Log river water depth and river bed elevation results in the shift log. ________

(5) Revise STP 3.0.0-01, Instrument Checks, to document final current sand gate elevation. ________

PROBABLE ANNUNCIATORS

None

PROBABLE INDICATIONS

River bed elevation > 726 feet

River bed elevation is determined to be less than or equal to (≤) 0.5 feet from the current sand gate elevation

IMMEDIATE ACTIONS
1. None

AUTOMATIC ACTIONS

– None

FOLLOW-UP ACTIONS

NOTE

River water elevation is measured via LR-2901, Computer Point M010.V, and/or STP 3.0.0-01. River water depth is determined utilizing 1H239 River Depth Measuring Crane per OI 410 or during surveillance testing. River bed elevation is determined by subtracting the 1H239 measured river water depth from river water elevation. River bedrock elevation ranges from 722' to 723'. The difference between river bed elevation and river bedrock elevation is considered the sand bed height. River bed elevation and sand bed elevation are synonymous with each other.

Current sand gate elevation is listed in STP 3.0.0-01.

Actions taken in this section are to be coordinated with the requirements of the High River Bed Elevation section of this AOP.

1. **IF** River Water Depth is less than (<) 1.5' above the Sand Gate elevation, **THEN** update the Online Risk Monitor by using the Review/Change Environmental Variables button to adjust Ultimate heat Sink to High Risk. ________

2. **IF** river water depth is determined to be less than or equal to (≤) 2' utilizing 1H239, **THEN** comply with Technical Specification SR 3.7.2.3 and SR 3.7.2.5. ________

3. **IF** river water depth is determined to be less (<) 2.5' utilizing 1H239,

   <u>OR</u>

   river water elevation is determined to be less than (<) 2.5' above the current sand gate elevation

   THEN perform the following

   a. Determine river water depth and river bed elevation daily using 1H239 per OI 410 River Water Depth section. Log river water depth and river bed elevation results in the shift log. ________

   b. **IF** river bed elevation is less than or equal to (≤) 725' elevation, **THEN** the OSM and System Engineering shall evaluate lowering the sand gate elevation to maintain river water elevation greater than or equal to (≥) 2.5' above the sand gate elevation. Sand gates may be lowered per Step 4. ________

FOLLOW-UP ACTIONS (continued)

c. **IF** river bed elevation is greater than (>) 725' elevation, **THEN** task and schedule the following prior to lowering the sand gates: ______

(1) Iowa Vane Excavation (Model WO 40158555) ______

- Schedule excavation to begin prior to river depth less than (<) 2.5' <u>OR</u> river level less than (<) 2.5' from the top of the gate.
- Excavate Iowa Vane field and river channel downstream of intake structure for a minimum of 100 yards.
- Excavate Iowa Vane field to river bottom (722' to 723') per mapping.
- Monitor screen differential pressure during excavation.

NOTE

Dive Work Orders are to be performed following completion of excavation. Inspection/cleaning includes the Forebay up to the river side of Lite Locs. Removal of Lite Locs and inspection/cleaning to Traveling Screen and RWS pump pits to be completed based on Forebay results and OSM direction. ______

(2) Diving inspection and cleaning work orders: ______

- 'A' side Intake structure forebay (Model WO 1374681) or 'B' side Intake structure forebay (Model WO 1375828) as applicable.
- 'A' side RWS pump pit (Model WO 1374679) and 'B' side RWS pump pit (Model WO 1374680) per OSM direction.

(3) Following excavation and diving, determine river water depth and river bed elevation using 1H239 per OI 410 River Water Depth section. Log river water depth and river bed elevation results in the shift log. ______

(4) Following excavation and diving, return to Step(s) 2 and/or 3 and take actions as applicable. ______

FOLLOW-UP ACTIONS (continued)

NOTE

Base sand gate elevation is 728.0'. Fully raised sand gate elevation is 730'. Fully lowered elevation is 723.5'. Both sand gates are to be maintained at the same elevation.

Raising sand gates will impact water level at the intake structure.

Divers must inspect and clean (if necessary) the intake structure forebay in the vicinity of the Sand Gates to verify the gates are clear prior to lowering the gates back to 'base' elevation.

Sand Gate electrical controls are non-functional. Sand Gates must be raised and lowered manually using manual ratchet tools.

Sand loading on the river side of the Sand Gates may impede or prevent gate movement.

The OSM and System Engineering shall consider the requirements of SR 3.7.2.3 and SR 3.7.2.5 for maintaining greater than 1' water level above the sand gate elevation when proposing to lower the sand gates.

4. **IF** desired to lower sand gate elevation per Step 3.b, **THEN** coordinate with the Work Week Manager, System Engineering, and Maintenance to perform the following as applicable:

NOTE

The area beneath the sand gates should be inspected and cleaned as necessary to remove sand accumulation that may prevent sand gate movement prior to attempting to lower the sand gates.

CAUTION

If sand gates are raised, sand may build up underneath the gates preventing the gates from being lowered. This could result in a loss of river water supply if river level drops below the top of the gates. To preclude this, the sand gates should be kept at least 1 foot below river water level as monitored in STP 3.0.0-01. **{C002}**

a. **IF** not already completed, **THEN** task and schedule Model WO 1374681 **OR** Model WO 1375828 for diver inspection/cleaning of the Intake Structure forebay.

b. Manually lower gates to desired elevation per OI 410 maintaining greater than or equal to (≥) 1' above river bed elevation (minimum sand gate elevation is 723.5').

FOLLOW-UP ACTIONS (continued)

c. Verify sand gate elevation using gate dial indicator, gate height measuring device, or divers. ________

d. Determine river water depth and river bed elevation utilizing 1H239 per OI 410, River Water Depth Section. Log river water depth and river bed elevation results in the shift log. ________

e. Revise STP 3.0.0-01, Instrument Checks, to document final current sand gate elevation. ________

f. Continue to perform applicable actions per Step(s) 2 and/or 3 until the conditions of 4.g are met. ________

g. IF river water depth is determined to be greater than or equal to (≥) 2.5' utilizing 1H239,

AND

IF river water elevation is determined to be greater than or equal to (≥) 732.5' to ensure greater than 2.5' is maintained above the current sand gate elevation

THEN the sand gates may be returned to 728' by performing the following:

(1) Verify all excavation, inspection and or cleaning as required by Step 3 and or Step 4.a is completed as necessary. ________

(2) Manually raise sand gates per OI 410 to base elevation of 728' or as determined by System Engineering. ________

(3) Verify final sand gate elevation using gate dial indicator, gate height measuring device, or divers. ________

(4) Determine river water depth and river bed elevation utilizing 1H239 per OI 410, River Water Depth Section. Log river water depth and river bed elevation results in the shift log. ________

(5) Revise STP 3.0.0-01, Instrument Checks, to document final current sand gate elevation. ________

5. **WHEN** River Water Depth is more than (>) 1.5' above the Sand Gate elevation, **THEN** update the Online Risk Monitor by using the Review/Change Environmental Variables button to adjust Ultimate heat Sink to Normal Risk. ________

LOW RIVER WATER DEPTH

PROBABLE ANNUNCIATORS

None

PROBABLE INDICATIONS

River water depth ≤ 2.5 feet

APPENDIX 1

INFORMATION

Below is a table of approximate times before the ESW/RHRSW Wet Pits and Stilling Basin empty on a complete loss of River Water supply. Times are based on an initial pit level of 28 feet (Reference CAL-M93-078).

Pumps Running	1 ESW	2 ESW	1 RHRSW 1 ESW	1 RHRSW 2 ESW	2 RHRSW 2 ESW
GPM Flow	1200	2400	3600	4800	7200
Minutes	92	46	31	23	15

River Water Supply Pumps can be started/restarted when the pump's respective white "START PERMISSIVE" indicating light is on. This light will be on when the following conditions are met:

- Control Power is available
- No existing undervoltage condition on respective Essential Bus
- Pump restart timer has timed out (approximately 2 minutes)
- Pump control switch is positioned to either AUTO or START

References

6. Technical Specifications
7. OI 410, River Water Supply System
8. OI 408, Well Water System
9. OI 411, General Service Water System
10. OI 416, RHR Service Water System
11. OI 442, Circ Water System
12. OI 454, Emergency Service Water System
13. P&ID M129, River Water Supply System Intake Structure
14. P&ID M146, Service Water System Pumphouse
15. Service Water Systems, Bechtel Drawing No. 7884-E-111<13, 13A>
16. 4160V and 480V System Control and Protection, Bechtel Drawing No. 7884-E-104<25, 26>
17. Bechtel Drawing No. 7884-APED-B21(3A)
18. Bechtel Drawing No. 7884-APED-B21-18(3A)NI
19. P&ID M15, Equipment Location Pumphouse Plans and Elevations
20. Architectural No. A-78, Pumphouse Plans and Elevations
21. Civil No. C-684, Pumphouse Conc. Floor Plans at 761'-0" and 747'-6"
22. DCP 1496, River Water Pump Restart Logic Mod
23. CAL-M93-078, RHRSW/ESW Pit Pumpdown Times
24. AR 95-1478, AR 95-2070-15, AR 18569, AR 23595
25. NG-94-4671, NG-95-2864

Commitment Items

26. **C001** - INPO SOER 07-2 (recommendation 5a) regarding 'Intake Cooling Water Blockage'
27. **C002** - Licensing CTS No. 199005110204, NRC Inspection Report 90003

ABNORMAL OPERATING PROCEDURE

AOP 410

LOSS OF RIVER WATER SUPPLY/HIGH RIVER BED ELEVATION/LOW RIVER WATER DEPTH

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: ____________________ / ____________________________ Date: __________
Print Signature

Reviewed By: ____________________ / ____________________________ Date: __________
Print Signature

PROCEDURE APPROVAL

Approved By ____________________ / ____________________________ Date: __________
Print Signature

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 a 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

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

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

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.

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.

Drywe			
	< 6%		(3)
	≥ 6% or unknown	(1)	

H-2

Torus	
	< 6%
	≥ 6% or unknown

H-6

1

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

IF........ offsite release r
(Detail D),
THEN...vent and purge
- ☛ OK to defe
- ☛ If pneuma
Venting th
Pneumati
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   - IF........ t
     THEN...
   - IF........ t
     AND.... t
     THEN...
2. IF........ the p
   THEN...purg
   (SEF
3. Stop the ven
   steps) when:
   - Hydroge
     OR
   - Offsite re

H-7

2

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

IF........ offsite release r
Emergency Lev
OR...... RPV water leve
THEN...vent and purge
- ☛ OK to defe
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2. IF........ the p
   THEN...purg
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3. Stop the ven
   steps) when:
   - Hydroge
     the torus
     OR
   - Hydroge

10

pumps to
er level
Operating

Temperature	**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

	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			

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).

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.

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

10

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er level
perating

Temperature

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

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			

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.
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<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

Maximize drywell cooling.

- If necessary, bypass drywell cooling isolation and fan speed interlock (Defeat 4).

DW/T-3 ☐

IF	THEN
Drywell pressure drops below **2.0 psig**	Verify containment sprays isolate.

BEFORE

drywell temperature reaches **280°F**

DW/T-4 ☐

EOP 1 ①

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).

DW/T-5 ☐

5

WAIT UNTIL

drywell temperature cannot be restored and maintained below **340°F**

DW/T-6

IF

Torus pressure drops below **2.0 psi**

Drywell pressure drops below **2.0 ps**

Primary containment pressure reduc required to:

- Restore and maintain adequate c
OR
- Reduce the total offsite radiation

PC/P-3 ☐

to

Initiate to
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- OK
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leve
belo

PC/P-4 ☐

PC/P-5 ☐

IF to

10

ɔumps to
er level
ɔperating

Temperature

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

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			

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

FOLLOW-UP ACTIONS (continued)

NOTE

River Water Level is required to be recorded hourly and verified to be < 757 feet whenever river level is >753 feet, IAW TLCO 3.7.1.

9. **IF** river water level reaches **753'** **THEN** Commence recording river water level hourly.

10. Pre-stage equipment in the location per Attachment 3 for the Pump House, Turbine Building, Control Building, Reactor and Recombiner Building, and Radwaste/LLRPSF

CAUTION

Deenergizing MCC 1B9106 [1B2106] will cause RWS Screen Wash Pump 1P-112A[B] to be inoperable. Refer to applicable Tech. Spec. Sections.

NOTE

Once level is greater than 754’, the only method to get into the Intake Structure without walking through water is to remove the louver on the Intake Air Supply and use a boat to get to the building.

11. Prior to river level reaching **754'**, open the following breakers to prevent energizing electrical equipment in the Intake Structure that may become submerged. Other loads at the Intake are on the 2nd floor and will be deenergized later at higher river levels.

Breaker	Load
1B9106	Screen Wash Pump 1P-112A
1B9107	Traveling Screen Drive 1F-36A
1B9108	Screen Wash Control Panel 1C-154A
1B9109	Radial Sand and Side Gate Hoist 1H-26A
1B9111	Intake Structure 480VAC Power Receptacles
1B9112	Screen Wash Nozzle Shutoff MO-2902
1B2106	Screen Wash Pump 1P-112B
1B2107	Traveling Screen Drive 1F-36B
1B2108	Screen Wash Control Panel 1C-154B
1B2109	Intake Structure Trash Rake 1S-83
1B2110	Intake Structure 480VAC Power Receptacles
1B2111	Radial Sand and Side Gate Hoist 1H-26B
1B2112	Screen Wash Nozzle Shutoff MO-2903
1B2114	Instrument Enclosure 1C412

FOLLOW-UP ACTIONS (continued)

12. When river level reaches **756'**, at the Pump House, Turbine Building, Control Building, Reactor and Recombiner Building, and Radwaste/LLRPSF implement action per Attachment 3 to pump water/monitor levels that may leak into these buildings in the future if river level continues to rise. ________

13. **IF** it is determined that the flood level will reach **757'** **OR** does reach **757'** **THEN**

a. Shut down the plant to Cold Shutdown ________

b. Deenergize equipment not required to shutdown the plant or for personal safety or for security or fuel pool cooling, at the following: ________

(1) Cooling towers
(2) Administration Bldg.
(3) Pump House
(4) Low Level Radwaste Process Storage Facility (LLRPSF) Building
(5) Machine Shop
(6) Offgas Retention Facility
(7) Security Bldg
(8) TSC Bldg
(9) Data Acquisition Center
(10) Badging Center
(11) Electric Shop
(12) Construction support Center
(13) West/East Warehouse
(14) Fabrication Shop
(15) Sewage Treatment Plant
(16) Shooting Range
(17) Barn

c. Refer to EPIP 1.1 for EAL assessment. ________

CAUTION

Major 4160V loads should be started at approximately **10** second intervals to avoid overloading the diesel generator.

Diesel generator load should be monitored at 1C08 as bus loads are added.

____

d. Start both SBDGs and transfer the 4160VAC Essential Busses 1A3 and 1A4 to the SBDGs per OI 304.2, Section 7.6. ________

ABNORMAL OPERATING PROCEDURE

AOP 410

LOSS OF RIVER WATER SUPPLY/HIGH RIVER BED ELEVATION/LOW RIVER WATER DEPTH

Usage Level
REFERENCE

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:

NOTE

Refer to EPIP 1.1 for EAL ASSESSMENT.

NOTE

The portions of this procedure that minimize river make-up flow and maximizes make-up flow from other sources may be used as necessary in the event of an intrusion of excess foreign material from the Cedar River and/or Intake Structure without a Loss of River Water Supply.

IMMEDIATE ACTIONS

1. None

AUTOMATIC ACTIONS

– CV-4914 and CV-4915 open and CV-4910A and CV-4910B close on low level in the ESW/RHRSW wet pits

FOLLOW-UP ACTIONS

1. Establish critical parameter monitoring of Circ Pit Level and ESW/RHRSW Pit Level, as priorities allow. ________

2.	**IF**	power is available **AND** annunciator 1C06A (A-1[2]) "A"["B"] RWS PIT LO LEVEL is not on	**THEN**	attempt to start standby pumps in both RWS Subsystems as necessary to restore needed makeup flow.	________
3.	**IF**	standby pumps did not start	**THEN**	attempt to restart the tripped River Water Pumps.	________
4.	**IF**	power is not available	**THEN**	attempt to restore power from 1C08 **AND** attempt to restart pumps.	________
5.	**IF**	no RWS pumps can be started  	**THEN**	Reduce recirc flow to 39 Mlbm/hr in accordance with IPOI 4 **AND** manually scram the reactor.	________
6.	**IF**	offsite power was lost **AND** River Water Supply pumps fail to start from the diesel	**THEN**	Attempt to start pumps manually, that have their start permissive light on. **AND** restore RWS makeup to the stilling basin.	________

7. Update the Online Risk Monitor for the status of River Water Supply Pumps. ________

8. Maximize Well Water flow for makeup to the Circ Pit, while maintaining ≤ 170 psig well water system pressure at the main plant BEECO Backflow Preventer. ________

9. Close the Circ Water Inlet to Blowdown Line valve MO-4253 to maintain Circ Water Pit inventory. ________

10. Secure Cooling Tower Fans as allowable ________

11. Secure one circ water pump and one cooling tower as soon as possible. ________

FOLLOW-UP ACTIONS (continued)

12. Send an operator to the Intake Structure to verify alarms and check RWS pump breaker condition. ________

> **NOTE**
>
> The RHRSW Pumps should be kept in operation to support use of the RHR System as directed by EOPs (Torus Cooling/Shutdown Cooling). The RHRSW Pumps may be secured when it is determined that ESW/RHRSW pit level cannot be maintained above 4 feet. Securing RHRSW Pumps prior to reaching 4 feet in the pits will preserve the ESW Supply to the SBDGs.

____

13. Minimize use of water from the RHRSW/ESW pit as follows:

    a. Secure RHRSW pumps unless required to support operation of the RHR System ________

    b. Shutdown any SBDG not required to ensure one Essential Bus is energized and/or required to ensure adequate core cooling. ________

        (1) Verify SBDG Cooling Valves CV-2080 and CV-2081 close when the respective SBDG is secured. ________

    c. Verify Well Water is available for cooling the operating Control Building Chiller and then secure ESW to the Control Building Chillers by unlocking and closing V-13-122 and V-13-125 on the Reactor Building 812' level. ________

14. Minimize heat addition to the Torus (Reliefs, HPCI, RCIC). ________

15. Use the Turbine Bypass valves and/or Bypass Jack for Reactor pressure control and Reactor cooldown and continue to bleed steam to the Main Condenser for as long as possible. ________

16. At 1C15 and 1C17, place the HI COND BACKPRESS BYPASS switches in BYPASS. ________

17. Notify Security at 7254 prior to opening Pumphouse doors to arrange for required Security compensatory measures. ________

FOLLOW-UP ACTIONS (continued)

18. Establish makeup to the ESW/RHRSW Wet Pits from the Fire System using 2-1/2" and/or 5" hoses as follows: **{C001}** ________

    a. Using 2-1/2" hoses (N/A if not used):

    (1) Notify Mechanical Maintenance to remove the cover from outside hose head (octopus head). ________

    (2) Obtain 2-1/2" hoses from the warehouse and fire brigade trailer, and rig as many as possible (8 preferred) from the octopus head to the stilling basin. ________

    (3) Lash the hoses together with rope to prevent hose whip when the lines are charged. ________

    b. Using 5" hoses (N/A if not used):

    (1) Obtain 5" hoses from the B5b hose trailer. ________

    (2) Connect a 5" hose to any of the following fire hydrants as required: ________

    - FH-1 located east of the Turbine Building
    - FH-2 located southeast of the Turbine Building
    - FH-7 located northeast of the Turbine Building

    (3) Rig 5" hoses as needed to the Stilling Basin. ________

    (4) Lash the hoses together with rope to prevent hose whip when the lines are charged. ________

    c. When directed by the CRS, start 1P-48 or 1P-49 and valve in hoses as necessary to maintain RHRSW/ESW pit level. ________

    d. Monitor RHRSW/ESW pit level at 1C29, computer points B279 and B280, or on group display AOP 410. ________

FOLLOW-UP ACTIONS (continued)

19. Establish makeup to the RHRSW/ESW pits from GSW as follows:

    a. Verify at least one GSW pump is running. ________

    b. At 1C452 (located in the B RHRSW/ESW pump room) open the following valves with the appropriate handswitch: ________

CV-8035A	A RHRSW/ESW WET PIT CHLORINE INJECTION ISOLATION	HS-8035A
CV-8035B	B RHRSW/ESW WET PIT CHLORINE INJECTION ISOLATION	HS-8035B
CV-8034	RHRSW/ESW DILUTION WATER SUPPLY VALVE	HS-8034

    c. In the CHLORINE BOOSTER PUMP ROOM, note and record the position of V-80-154, then fully open V-80-154 GSW Dilution Water Supply Balancing valve using a wrench from the tool board. ________

NOTE

NPSH requirement is 8 feet for the Circ Water Pumps. NPSH requirement is 4 feet for the ESW, RHRSW, and GSW pumps. ________

20. Monitor the Circ Water Pit level at Computer Point F092 and secure Circ Water Pumps if level cannot be maintained or restored greater than 8 feet and GSW pumps as necessary to prevent cavitation. ________

21. Monitor RHRSW/ESW Pit level and secure RHRSW and ESW pumps as necessary to prevent cavitation. ________

22. When the status of each River Water Supply pump is known and there is at least one operable pump in each RWS loop, select the operable pumps on HSS-2911A and B to be the pumps to auto restart on the diesel. ________

23. Comply with Technical Specifications for River Water Supply. ________

24. When River Water Supply pump operation is restored, return system to normal operation per OI 410. ________

25. Update the Online Risk Monitor for the status of River Water Supply Pumps. ________

FOLLOW-UP ACTIONS (continued)

26. Open and Lock open the following valves:

- V-13-122 ESW Loop A Return Header Isolation ________
- V-13-125 ESW Loop B Return Header Isolation ________

27. Independently verify the following valves are locked open:

- V-13-122 ESW Loop A Return Header Isolation ________ IV
- V-13-125 ESW Loop B Return Header Isolation ________ IV

PROBABLE ANNUNCIATORS

1C06A,	A-1	"A" RWS PIT LO LEVEL
	A-2	"B" RWS PIT LO LEVEL
	A-3	"B" RWS PUMP 1P-117B TRIP
	A-4	"D" RWS PUMP 1P-117D TRIP
	B-1	"A" RWS PUMP 1P-117A TRIP
	B-2	"C" RWS PUMP 1P-117C TRIP
	B-5	"A" COOLING TOWER BASIN HI/LO LEVEL
	B-6	"B" COOLING TOWER BASIN HI/LO LEVEL
	D-1	"A" RHRSW/ESW PIT LO LEVEL
	D-2	"B" RHRSW/ESW PIT LO LEVEL
	D-11	CIRC WATER PIT LO LEVEL

PROBABLE INDICATIONS

1C06

– River Water makeup flow stopped at FR-4916 and FR-4917 or FI-4916 and FI-4917

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.

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.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
OR	AND	
Reactor Coolant specific activity > 2.0 μCi/gm DOSE EQUIVALENT I-131.	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.

FIGURE F-5
DAEC TELEPHONE SYSTEMS

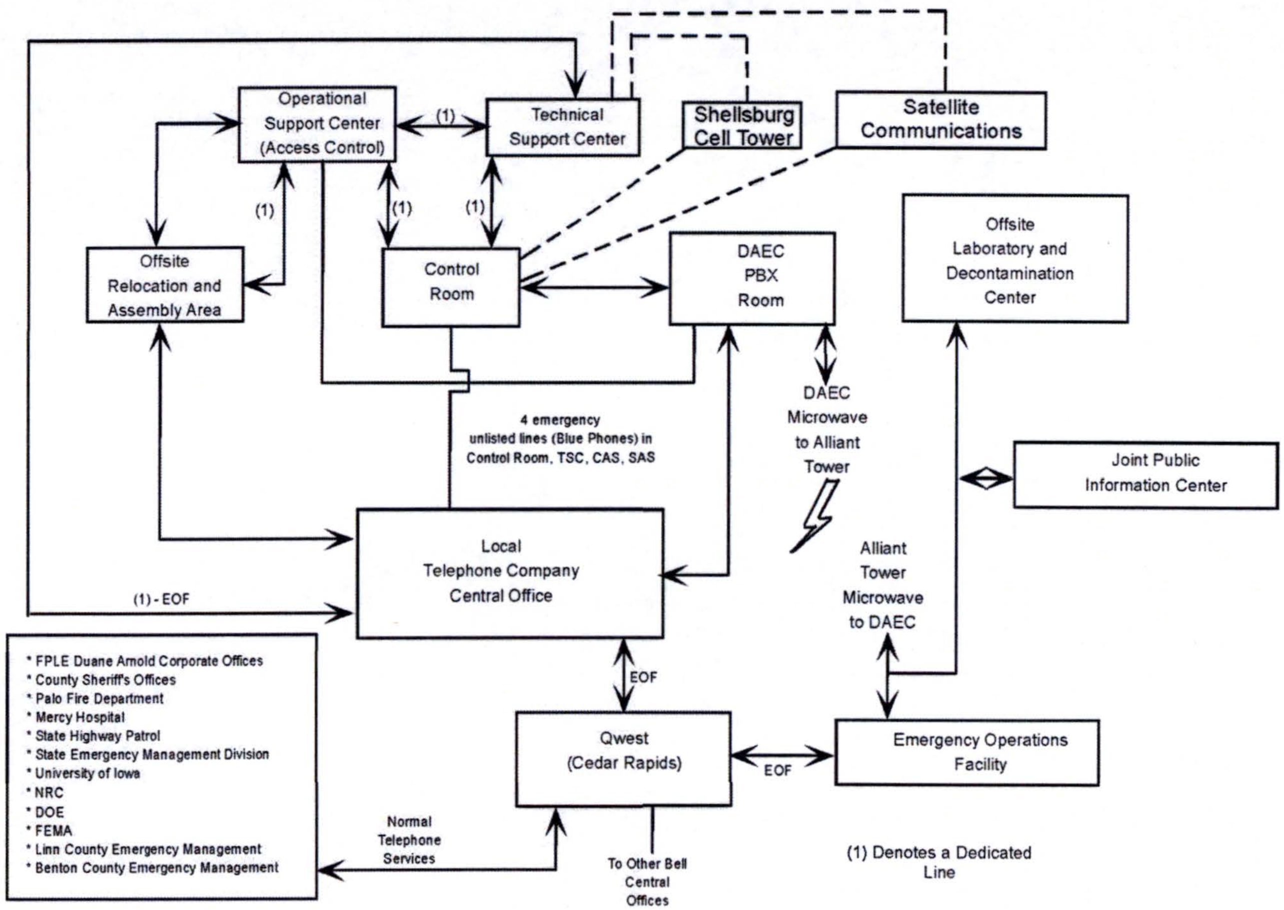


FIGURE F-6
FEDERAL TELEPHONE SYSTEM (FTS-2001)

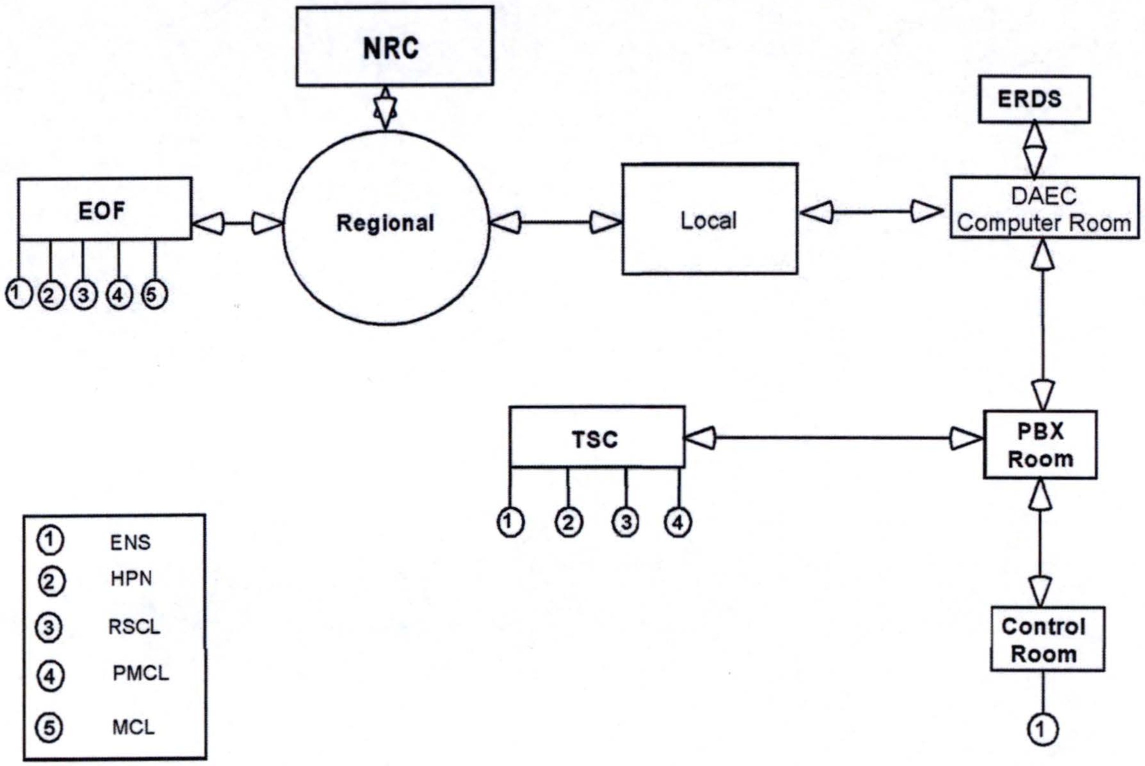
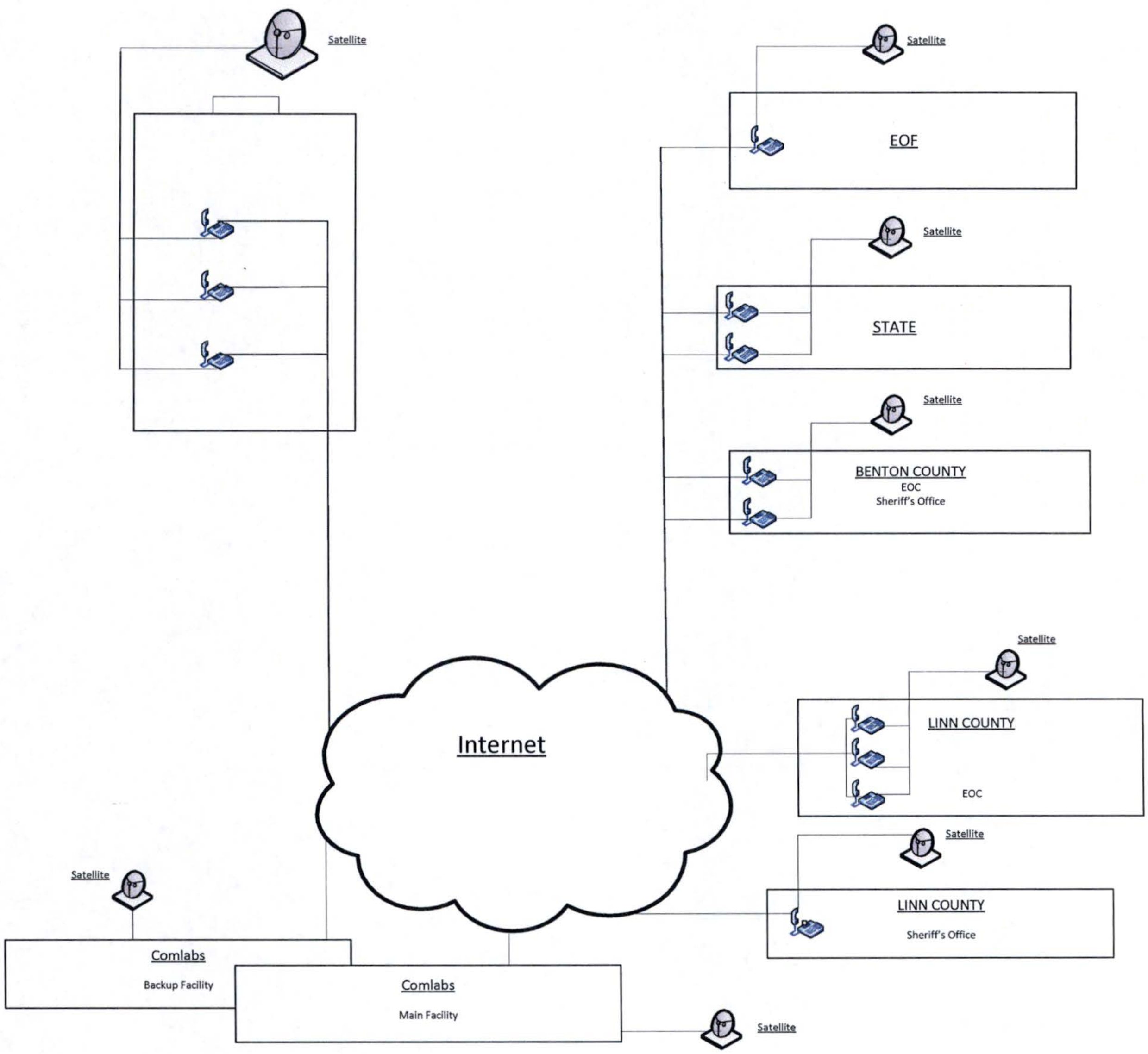


FIGURE F-7
ALL-CALL TELEPHONE SYSTEM



FOLLOW-UP ACTIONS (continued)

NOTE

River Water Level is required to be recorded hourly and verified to be < 757 feet whenever river level is >753 feet, IAW TLCO 3.7.1.

9. **IF** river water level reaches **753'** **THEN** Commence recording river water level hourly.

10. Pre-stage equipment in the location per Attachment 3 for the Pump House, Turbine Building, Control Building, Reactor and Recombiner Building, and Radwaste/LLRPSF

CAUTION

Deenergizing MCC 1B9106 [1B2106] will cause RWS Screen Wash Pump 1P-112A[B] to be inoperable. Refer to applicable Tech. Spec. Sections.

NOTE

Once level is greater than 754', the only method to get into the Intake Structure without walking through water is to remove the louver on the Intake Air Supply and use a boat to get to the building.

11. Prior to river level reaching **754'**, open the following breakers to prevent energizing electrical equipment in the Intake Structure that may become submerged. Other loads at the Intake are on the 2nd floor and will be deenergized later at higher river levels.

Breaker	Load
1B9106	Screen Wash Pump 1P-112A
1B9107	Traveling Screen Drive 1F-36A
1B9108	Screen Wash Control Panel 1C-154A
1B9109	Radial Sand and Side Gate Hoist 1H-26A
1B9111	Intake Structure 480VAC Power Receptacles
1B9112	Screen Wash Nozzle Shutoff MO-2902
1B2106	Screen Wash Pump 1P-112B
1B2107	Traveling Screen Drive 1F-36B
1B2108	Screen Wash Control Panel 1C-154B
1B2109	Intake Structure Trash Rake 1S-83
1B2110	Intake Structure 480VAC Power Receptacles
1B2111	Radial Sand and Side Gate Hoist 1H-26B
1B2112	Screen Wash Nozzle Shutoff MO-2903
1B2114	Instrument Enclosure 1C412

FOLLOW-UP ACTIONS (continued)

12. When river level reaches **756'**, at the Pump House, Turbine Building, Control Building, Reactor and Recombiner Building, and Radwaste/LLRPSF implement action per Attachment 3 to pump water/monitor levels that may leak into these buildings in the future if river level continues to rise. ________

13. **IF** it is determined that the flood level will reach **757'** **OR** does reach **757'** **THEN**

    a. Shut down the plant to Cold Shutdown ________

    b. Deenergize equipment not required to shutdown the plant or for personal safety or for security or fuel pool cooling, at the following: ________

    (1) Cooling towers
    (2) Administration Bldg.
    (3) Pump House
    (4) Low Level Radwaste Process Storage Facility (LLRPSF) Building
    (5) Machine Shop
    (6) Offgas Retention Facility
    (7) Security Bldg
    (8) TSC Bldg
    (9) Data Acquisition Center
    (10) Badging Center
    (11) Electric Shop
    (12) Construction support Center
    (13) West/East Warehouse
    (14) Fabrication Shop
    (15) Sewage Treatment Plant
    (16) Shooting Range
    (17) Barn

    c. Refer to EPIP 1.1 for EAL assessment. ________

CAUTION

Major 4160V loads should be started at approximately **10** second intervals to avoid overloading the diesel generator.

Diesel generator load should be monitored at 1C08 as bus loads are added. ________

    d. Start both SBDGs and transfer the 4160VAC Essential Busses 1A3 and 1A4 to the SBDGs per OI 304.2, Section 7.6. ________

ABNORMAL OPERATING PROCEDURE

AOP 410

LOSS OF RIVER WATER SUPPLY/HIGH RIVER BED ELEVATION/LOW RIVER WATER DEPTH

Usage Level
REFERENCE

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:

NOTE

Refer to EPIP 1.1 for EAL ASSESSMENT.

NOTE

The portions of this procedure that minimize river make-up flow and maximizes make-up flow from other sources may be used as necessary in the event of an intrusion of excess foreign material from the Cedar River and/or Intake Structure without a Loss of River Water Supply.

IMMEDIATE ACTIONS

1. None

AUTOMATIC ACTIONS

- CV-4914 and CV-4915 open and CV-4910A and CV-4910B close on low level in the ESW/RHRSW wet pits

FOLLOW-UP ACTIONS

1. Establish critical parameter monitoring of Circ Pit Level and ESW/RHRSW Pit Level, as priorities allow. ________

2.	**IF**	power is available **AND** annunciator 1C06A (A-1[2]) "A"["B"] RWS PIT LO LEVEL is not on	**THEN**	attempt to start standby pumps in both RWS Subsystems as necessary to restore needed makeup flow.	________
3.	**IF**	standby pumps did not start	**THEN**	attempt to restart the tripped River Water Pumps.	________
4.	**IF**	power is not available	**THEN**	attempt to restore power from 1C08 **AND** attempt to restart pumps.	________
5.	**IF**	no RWS pumps can be started 	**THEN**	Reduce recirc flow to 39 Mlbm/hr in accordance with IPOI 4 **AND** manually scram the reactor.	________
6.	**IF**	offsite power was lost **AND** River Water Supply pumps fail to start from the diesel	**THEN**	Attempt to start pumps manually, that have their start permissive light on. **AND** restore RWS makeup to the stilling basin.	________

7. Update the Online Risk Monitor for the status of River Water Supply Pumps. ________
8. Maximize Well Water flow for makeup to the Circ Pit, while maintaining ≤ 170 psig well water system pressure at the main plant BEECO Backflow Preventer. ________
9. Close the Circ Water Inlet to Blowdown Line valve MO-4253 to maintain Circ Water Pit inventory. ________
10. Secure Cooling Tower Fans as allowable ________
11. Secure one circ water pump and one cooling tower as soon as possible. ________

FOLLOW-UP ACTIONS (continued)

12. Send an operator to the Intake Structure to verify alarms and check RWS pump breaker condition. ________

NOTE ________

The RHRSW Pumps should be kept in operation to support use of the RHR System as directed by EOPs (Torus Cooling/Shutdown Cooling). The RHRSW Pumps may be secured when it is determined that ESW/RHRSW pit level cannot be maintained above 4 feet. Securing RHRSW Pumps prior to reaching 4 feet in the pits will preserve the ESW Supply to the SBDGs.

13. Minimize use of water from the RHRSW/ESW pit as follows:

    a. Secure RHRSW pumps unless required to support operation of the RHR System ________

    b. Shutdown any SBDG not required to ensure one Essential Bus is energized and/or required to ensure adequate core cooling. ________

        (1) Verify SBDG Cooling Valves CV-2080 and CV-2081 close when the respective SBDG is secured. ________

    c. Verify Well Water is available for cooling the operating Control Building Chiller and then secure ESW to the Control Building Chillers by unlocking and closing V-13-122 and V-13-125 on the Reactor Building 812' level. ________

14. Minimize heat addition to the Torus (Reliefs, HPCI, RCIC). ________

15. Use the Turbine Bypass valves and/or Bypass Jack for Reactor pressure control and Reactor cooldown and continue to bleed steam to the Main Condenser for as long as possible. ________

16. At 1C15 and 1C17, place the HI COND BACKPRESS BYPASS switches in BYPASS. ________

17. Notify Security at 7254 prior to opening Pumphouse doors to arrange for required Security compensatory measures. ________

FOLLOW-UP ACTIONS (continued)

18. Establish makeup to the ESW/RHRSW Wet Pits from the Fire System using 2-1/2" and/or 5" hoses as follows: ________
{C001}

a. Using 2-1/2" hoses (N/A if not used):

(1) Notify Mechanical Maintenance to remove the cover from outside hose head (octopus head). ________

(2) Obtain 2-1/2" hoses from the warehouse and fire brigade trailer, and rig as many as possible (8 preferred) from the octopus head to the stilling basin. ________

(3) Lash the hoses together with rope to prevent hose whip when the lines are charged. ________

b. Using 5" hoses (N/A if not used):

(1) Obtain 5" hoses from the B5b hose trailer. ________

(2) Connect a 5" hose to any of the following fire hydrants as required: ________

- FH-1 located east of the Turbine Building
- FH-2 located southeast of the Turbine Building
- FH-7 located northeast of the Turbine Building

(3) Rig 5" hoses as needed to the Stilling Basin. ________

(4) Lash the hoses together with rope to prevent hose whip when the lines are charged. ________

c. When directed by the CRS, start 1P-48 or 1P-49 and valve in hoses as necessary to maintain RHRSW/ESW pit level. ________

d. Monitor RHRSW/ESW pit level at 1C29, computer points B279 and B280, or on group display AOP 410. ________

FOLLOW-UP ACTIONS (continued)

19. Establish makeup to the RHRSW/ESW pits from GSW as follows:

    a. Verify at least one GSW pump is running. ________

    b. At 1C452 (located in the B RHRSW/ESW pump room) open the following valves with the appropriate handswitch: ________

CV-8035A	A RHRSW/ESW WET PIT CHLORINE INJECTION ISOLATION	HS-8035A
CV-8035B	B RHRSW/ESW WET PIT CHLORINE INJECTION ISOLATION	HS-8035B
CV-8034	RHRSW/ESW DILUTION WATER SUPPLY VALVE	HS-8034

    c. In the CHLORINE BOOSTER PUMP ROOM, note and record the position of V-80-154, then fully open V-80-154 GSW Dilution Water Supply Balancing valve using a wrench from the tool board. ________

NOTE

NPSH requirement is 8 feet for the Circ Water Pumps. NPSH requirement is 4 feet for the ESW, RHRSW, and GSW pumps. ________

20. Monitor the Circ Water Pit level at Computer Point F092 and secure Circ Water Pumps if level cannot be maintained or restored greater than 8 feet and GSW pumps as necessary to prevent cavitation. ________

21. Monitor RHRSW/ESW Pit level and secure RHRSW and ESW pumps as necessary to prevent cavitation. ________

22. When the status of each River Water Supply pump is known and there is at least one operable pump in each RWS loop, select the operable pumps on HSS-2911A and B to be the pumps to auto restart on the diesel. ________

23. Comply with Technical Specifications for River Water Supply. ________

24. When River Water Supply pump operation is restored, return system to normal operation per OI 410. ________

25. Update the Online Risk Monitor for the status of River Water Supply Pumps. ________

FOLLOW-UP ACTIONS (continued)

26. Open and Lock open the following valves:

    - V-13-122 ESW Loop A Return Header Isolation ________
    - V-13-125 ESW Loop B Return Header Isolation ________

27. Independently verify the following valves are locked open:

    - V-13-122 ESW Loop A Return Header Isolation ________ IV
    - V-13-125 ESW Loop B Return Header Isolation ________ IV

PROBABLE ANNUNCIATORS

1C06A,	A-1	"A" RWS PIT LO LEVEL
	A-2	"B" RWS PIT LO LEVEL
	A-3	"B" RWS PUMP 1P-117B TRIP
	A-4	"D" RWS PUMP 1P-117D TRIP
	B-1	"A" RWS PUMP 1P-117A TRIP
	B-2	"C" RWS PUMP 1P-117C TRIP
	B-5	"A" COOLING TOWER BASIN HI/LO LEVEL
	B-6	"B" COOLING TOWER BASIN HI/LO LEVEL
	D-1	"A" RHRSW/ESW PIT LO LEVEL
	D-2	"B" RHRSW/ESW PIT LO LEVEL
	D-11	CIRC WATER PIT LO LEVEL

PROBABLE INDICATIONS

<u>1C06</u>

– River Water makeup flow stopped at FR-4916 and FR-4917 or FI-4916 and FI-4917

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 a 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

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

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

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.

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

Maximize drywell cooling.

☛ If necessary, bypass drywell cooling isolation and fan speed interlock (Defeat 4).

DW/T-3 ☐

IF	THEN
Drywell pressure drops below **2.0 psig**	Verify containment sprays isolate.

BEFORE

drywell temperature reaches **280°F**

DW/T-4 ☐

EOP 1 ①

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 <u>only</u> pumps <u>not</u> required for adequate core cooling (Table 9).

DW/T-5 ☐

5

WAIT UNTIL

drywell temperature <u>cannot</u> be restored and maintained below **340°F**

DW/T-6

IF

Torus pressure drops below **2.0 psi**

Drywell pressure drops below **2.0 ps**

Primary containment pressure reduc required to:

- Restore and maintain adequate c
  OR
- Reduce the total offsite radiation

PC/P-3 ☐

to

Initiate to required f

☛ OK con leve bel

PC/P-4 ☐

PC/P-5 ☐

IF to

FOLLOW-UP ACTIONS (continued)

18. Direct an operator and electricians (if available) to perform Attachment 10 Alternate AC power to 125VDC and 250VDC chargers. ________

NOTE

The inverters will automatically trip at 105 VDC decreasing.

____

19. After one hour has elapsed, dispatch an operator to implement Attachment 13, Load Shedding to Preserve Station Batteries (from AOP 301.1 hanging file). Attachment 13 is to be completed within two hours of the SBO event. ________

20. **IF** a SBDG is available for operation, **THEN** restore power to its essential bus per Restoration of Standby Diesel Generator Power section. ________

21. **WHEN** the DAEC Switchyard inspection has been completed, **THEN** restore power to the switchyard per Restoration of Offsite Power section. ________

22. **WHEN** sufficient offsite power becomes available, **THEN** restore power to the non-essential buses per AOP 304.1. ________

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 480 V 480 V	Essential Bus 1A3 Load Centers 1B3, 1B9 Motor Control Centers 1B32, 1B34	Essential Bus 1A4 Load Centers 1B4, 1B20 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.

ATTACHMENT 5

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST TSCR-166

DAEC EAL SCHEME WALLBOARDS
[FOR INFORMATION ONLY]

2 pages follow