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December 15, 2016

10 CFR 50.90

SBK-L-16196

Docket No. 50-443

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Seabrook Station

Response to Request for Supplemental Information Regarding License Amendment Request 15-02, Adoption of Emergency Action Level Schemes Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors"

References:

1. NextEra Energy Seabrook, LLC letter SBK-L-15120, "License Amendment Request 15-02, Adoption of Emergency Action Level Schemes Pursuant to NEI 99-01, Revision 6 'Development of Emergency Action Levels for Non-Passive Reactors'" February 27, 2016 (ML16068A128)

2. NRC letter "Seabrook Station, Unit No. 1 – Request for Additional Information Related to License Amendment Request to Adopt Emergency Action Level Schemes Pursuant to NEI 99-01, Revision 6 (CAC MF7439)," September 22, 2016 (ML16230A533)

3. NextEra Energy Seabrook, LLC letter SBK-L-16162, "Response to Request for Additional Information Regarding License Amendment Request 15-02, Adoption of Emergency Action Level Schemes Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors", October 27, 2016 (ML16302A414)

4. NRC e-mail "Need for Supplement to EAL license amendment" November 10, 2016 (ML16319A421)

In Reference 1 and supplemented by Reference 3, NextEra Energy Seabrook, LLC (NextEra) submitted a license amendment request (LAR) to revise the current EAL scheme to one based upon the Nuclear Energy Institute (NEI) document NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors".

In Reference 4, based on a teleconference held on November 7, 2016 to clarify NextEra's responses in Reference 3, the NRC staff requested that NextEra provide clarifications related to the responses provided for RAI-Seabrook-3 and RAI-Seabrook-5 in Reference 3. Additionally, in order to provide a clear licensing basis for the staff to reference, the NRC has requested a

AX45
NRR

clean version of the EALs be provided in this supplement. Enclosure 1 to this letter supplements NextEra's responses to RAI-Seabrook-3 and RAI-Seabrook-5. Enclosure 2 provides a markup of the proposed emergency action level revised with this supplement, which supersedes the corresponding markup in Reference 1. Enclosure 3 includes a clean copy of the Seabrook Station Emergency Action Levels- Initiating Conditions, Threshold Values, and Basis, and Enclosure 4 contains the table of NEI 99-01, Rev. 6, Deviations and Differences.

This supplement to LAR 15-02 does not alter the conclusion in Reference 1 that the changes do not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the changes.

No new or revised commitments are included in this letter.

Should you have any questions regarding this letter, please contact Mr. Kenneth Browne, Licensing Manager, at (603) 773-7932.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 15, 2016.

Sincerely,

Eric McCartney
Site Vice President
NextEra Energy Seabrook, LLC

Enclosures

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Enclosure 1 to SBK-L-16196

Response to Request for Supplemental Information Regarding License Amendment Request 15-02, Adoption of Emergency Action Level Schemes Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors"

Background

NextEra Energy Seabrook Station letter dated October 27, 2016 (SBK-L-16162) provided responses to the NRC staff's request for additional information (RAI) related to the license amendment request regarding revising the current EAL scheme to one based upon the Nuclear Energy Institute (NEI) document NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors". On November 7, 2016 NRC staff held a phone call with NextEra staff to discuss the responses to RAIs 3 and 5. NRC has requested that the clarifying information be provided to NRC in a supplemental letter. The information below provides the requested supplemental information as discussed during the November 7, 2016 phone call.

RAI-Seabrook-3 NRC Follow-up Question

During the call the NRC staff clarified that the question was directed towards the equipment's alarm setpoint and its tie to exceeding 2 times a release control limit. The NRC staff requests that NextEra supplement its response, as well as revise the wording in the EAL back to the current wording, as discussed during the call.

NextEra Response:

Per discussion during the November 7, 2016, phone call, EAL RU1(1) is revised back to the wording that is currently used for EAL AU1(1) in the existing Seabrook Station NEI 99-01 Rev. 4 EALs. EAL RU1(2) [NEI 99-01 Rev. 6 EAL AU1(2)] is deleted because it is redundant to EAL RU1(1). This is noted in the table of differences and deviations. EAL RU1(3) [NEI 99-01 Rev. 6 EAL AU1(3)] is re-designated EAL RU1(2) and is retained as written because it differs from the existing Seabrook Station NEI 99-01 Rev. 4 EAL AU1(2) in format only.

RAI-Seabrook-5 NRC Follow-up Question:

During the call, NextEra was able to clarify to the NRC staff that both SEPS are required by mentioning specific loading requirements in relation to the capacities of the equipment. Please include these values as discussed during the call.

NextEra Response:

Supplemental emergency power sysem (SEPS) Loading Calculation 9763-3-ED-00-02-F shows that the required load to maintain core cooling is greater than the capacity of one SEPS generator engine (2640 KW). Abnormal operating procedure OS1246.01, Loss of Offsite Power Plant Shutdown, contains instructions to restore power using SEPS. When SEPS is aligned to an emergency bus, OS1246.01 directs checking that the following equipment is loading:

- Service water pump
- Primary component cooling water pump
- Residual heat removal pump
- Charging pump

Calculation 9763-3-ED-00-02-F shows that the base load that is sequenced on once the emergency bus is energized and the sequencer finishes is approximately 1612 KW. Calculation 9763-3-ED-00-02-F identifies the following rated loads for the required equipment:

LOAD	Rated KW
Charging pump	554
Residual heat removal pump	343
Primary component cooling water pump	549
Service water pump	506
Total equipment load	1952
Total equipment load plus base load	3564

The total load of 3564 KW exceeds the capacity of one SEPS engine. The technical basis for EALs MG8, MG1, MS1, MA1 and CA2 is revised to add a sentence following the statement that says, "For power restoration from the SEPS both SEPS diesel generator sets must be functional." The added sentence says, "Both SEPS engines are required to power the emergency bus and equipment required to maintain core cooling."

NRC Additional Request:

In addition, in order to provide a clear licensing basis for the staff to reference both now and, if approved, in the future, please provide a clean version of the EALs in this supplement.

NextEra Response:

Clean version of all the EALs can be found in Enclosure 3- Clean Copy of Seabrook Station Emergency Action Levels – Initiating Conditions, Threshold Values and Basis. A clean version of NEI 99-01, Rev. 6, Deviations and Differences, can be found in Enclosure 4.

Enclosure 2 to SBK-L-16196

Markup of Affected Seabrook Station Emergency Action Levels – Initiating Conditions, Threshold Values and Basis

RU1

Notification of Unusual Event

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the ~~(site- specific effluent release controlling document)~~ ODCM limits for 60 minutes or longer.

Operating Mode Applicability: All

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

Notes:

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

(1) a. VALID ~~R~~reading on **ANYof the following** effluent ~~radiation~~ monitors greater than 2 times the ~~(site-specific effluent release controlling document) limits~~ value of the current high-alarm setpoint for 60 minutes or longer:

RM-6509-1 (WTT Disch)
RM-6521-1 (TB Sump)
RM-6519-1 (SG Blowdown)
RM-6473-1 (WT LIQ EFF)
RM-6528-4 (WRGM rate)

AND

b. The discharge flow to the environment is not isolated within 60 minutes.

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

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

OR

(~~3~~2) 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)~~ ODCM limits for 60 minutes or longer.

Basis:

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

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

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

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

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

EAL #1 - This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways. EAL #1 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the ODCM limit and releases are not terminated within 60 minutes. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

The discharge flowpaths associated with RM-6509-1, 6521-1, 6519-1, and 6473-1 have automatic and manual flow isolation capability. The EAL wording addresses a situation where a residual source term exists in a discharge flowpath AFTER the flowpath has been isolated, and the associated radiation monitor remains at values above 2 times the value of the current high-alarm setpoint. EAL 1.b ensures that the Initiating Condition (IC) intent of "to the environment" is met. The 60-minute assessment clock starts at the same time for both EAL 1.a and 1.b (i.e., clocks run concurrently). There must be a release to the environment (i.e., the flowpath can not be isolated) during the same period that a monitor value is greater than 2 times the value of the current high-alarm setpoint.

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

EAL #~~3~~2 - This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways ~~(e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.)~~.

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

Enclosure 3 to SBK-L-16196

Clean Copy of Seabrook Station Emergency Action Levels – Initiating Conditions, Threshold Values and Basis

SEABROOK STATION
EMERGENCY ACTION LEVELS

INITIATING CONDITIONS, THRESHOLD VALUES AND BASIS

TABLE OF CONTENTS

EMERGENCY ACTION LEVELS

1 REGULATORY BACKGROUND

1.1 OPERATING REACTORS

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

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

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

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

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 5, ISFSI ICs/EALs. IC EU1 covers the spectrum of credible natural and man-made events included within the scope of an ISFSI design.

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.

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 existing IC RA2, and new ICs RS2 and RG2. Associated EAL notes, bases and developer notes are also provided.

It is recommended that these EALs be implemented when the enhanced spent fuel pool level instrumentation is available for use.

The regulatory process that licensees follow to make changes to their emergency plan, including non-scheme changes to EALs, is 10 CFR 50.54(q). In accordance with this regulation, licensees are responsible for evaluating a proposed change and determining whether or not it results in a reduction in the effectiveness of the plan. As a result of the licensee's determination, the licensee will either make the change or submit it to the NRC for prior review and approval in accordance with 10 CFR 50.90.

1.4 ORGANIZATION AND PRESENTATION OF INFORMATION

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

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

1.5 IC AND EAL MODE APPLICABILITY

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

MODE APPLICABILITY MATRIX

	Category					
Mode	**R**	**C**	**E**	**F**	**H**	**M**
Power Operations	X		X	X	X	X
Startup	X		X	X	X	X
Hot Standby	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	

Operating Modes

Technical Specifications

TABLE 1.2

MODE		Reactivity Condition, K_{eff}	% Rated Thermal Power*	Average Coolant Temperature
1.	Power Operation	≥ 0.99	> 5%	≥ 350°F
2.	Startup	≥ 0.99	≤ 5%	≥ 350°F
3.	Hot Standby	< 0.99	0	≥ 350°F
4.	Hot Shutdown	< 0.99	0	350 °F > T_{avg} >200 °F
5.	Cold Shutdown	< 0.99	0	< 200 °F
6.	Refueling**	NA	0	< 140 °F
NA	Defueled	All fuel removed from the reactor vessel (full core offload during refueling or extended outage)		

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

1.6 BASIS DOCUMENT

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

- A site-specific Mode Applicability Matrix and description of operating modes (see Section 1.5).
- A discussion of the emergency classification and declaration process (see Section 2).
- Each Initiating Condition along with the associated EALs or fission product barrier thresholds, Operating Mode Applicability, Notes and Basis information (see Sections 3-8).
- A listing of acronyms and defined terms (see Appendices A and B, respectively).

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

Basis information should be readily available to be referenced, if necessary, by the Short Term Emergency Director/Site Emergency Director (STED/SED). For example, a copy of the basis document could be maintained in the appropriate emergency response facilities.

Because the information in a basis document can affect emergency classification decision-making (e.g., the STED/SED refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q).

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

The criteria/values used in several EALs and fission product barrier thresholds may be drawn from 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.

2 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

2.1 GENERAL CONSIDERATIONS

When making an emergency classification, the STED/SED 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, the STED/SED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

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

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded; 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.

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 scheme provides the STED/SED 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 STED/SED 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.

2.2 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.

2.3 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, 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, 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*.

2.4 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.

2.5 CLASSIFICATION OF IMMINENT CONDITIONS

Although EALs provide specific thresholds, the STED/SED 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 STED/SED, 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.

2.6 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.

2.7 CLASSIFICATION OF SHORT-LIVED EVENTS

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 EAL 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.

2.8 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. 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 STED/SED 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.

2.9 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.

3 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE. *Op. Modes: All*	**RS1** Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE. *Op. Modes: All*	**RA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE. *Op. Modes: All*	**RU1** Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer. *Op. Modes: All*
RG2 Spent fuel pool level cannot be restored to at least 1.5 ft (Level 3) for 60 minutes or longer. *Op. Modes: All*	**RS2** Spent fuel pool level at 1.5 ft. (Level 3). *Op. Modes: All*	**RA2** Significant lowering of water level above, or damage to, irradiated fuel. *Op. Modes: All*	**RU2** UNPLANNED loss of water level above irradiated fuel. *Op. Modes: All*
		RA3 Radiation levels that IMPEDE access to equipment necessary for normal plant operations, shutdown or cooldown. *Op. Modes: All*	

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: (1 or 2 or 3)

Notes:

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

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

Monitor	**Reading**	
RM-6528-4 (WRGM rate)	2.85E+8 uCi/sec	
	Time After Shutdown Reading	
	≤ 1 hr	**> 1 hr to ≤ 2 hrs**
RM-6481-1* (MSL A)	1310 mR/hr	1060 mR/hr
RM-6482-1* (MSL B)	1310 mR/hr	1060 mR/hr
RM-6482-2* (MSL C)	1310 mR/hr	1060 mR/hr
RM-6481-2* (MSL D)	1310 mR/hr	1060 mR/hr

* With release path to the environment from affected steam line, open ASDV or SRV, line is faulted, or open steam supply to 1-FW-P-37A.

OR

(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.

OR

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

Closed window dose rates greater than 1000 mR/hr expected to continue for 60 minutes or longer.
Analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for one hour of inhalation.

Basis:

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

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

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

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

ECL: General Emergency

Initiating Condition: Spent fuel pool level cannot be restored to at least1.5 ft. (Level 3) for 60 minutes or longer.

Operating Mode Applicability: All

Emergency Action Levels:

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

(1) Spent fuel pool level cannot be restored to at least 1.5 ft. above the fuel racks for 60 minutes or longer as indicated by SF-LI-2616 (MPCS computer point A4172) or SF-LI-2617 (MPCS computer point A4220).

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.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 12 ft. 3 in. above the top of the fuel racks (Level 2) and SFP level 1.5 ft. above the top of the fuel racks (Level 3).

The Spent Fuel Pool Instrumentation System (SFPIS) consists of two new independent level instrument channels (SF-L-2616 and SF-L-2617) in the Spent Fuel system. The SFPIS channels will be used to monitor spent fuel pool level during and following beyond design basis events that could challenge the capability to ensure optimum protection for the stored fuel assemblies in the pool.

Each channel is capable of measuring SFP level over a span from just above the top of the spent fuel racks to the normal SFP operating water level. The SFPIS will be monitored in accordance with Beyond Design Basis guidelines contained in FSGs for Extended Loss of AC Power and Alternate SFP Makeup and Cooling.

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.

RS1

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: (1 or 2 or 3)

Notes:

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

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

Monitor	Reading	
RM-6528-4 (WRGM rate)	2.85E+7 uCi/sec	
	Time After Shutdown Reading	
	≤ 1 hr	**> 1 hr to ≤ 2 hrs**
RM-6481-1* (MSL A)	130 mR/hr	100 mR/hr
RM-6482-1* (MSL B)	130 mR/hr	100 mR/hr
RM-6482-2* (MSL C)	130 mR/hr	100 mR/hr
RM-6481-2* (MSL D)	130 mR/hr	100 mR/hr

* With release path to the environment from affected steam line, open ASDV or SRV, line is faulted, or open steam supply to 1-FW-P-37A.

OR

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

OR

(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.

Basis:

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

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

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

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

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

ECL: Site Area Emergency

Initiating Condition: Spent fuel pool level at 1.5 ft.(Level 3)

Operating Mode Applicability: All

Emergency Action Levels:

(1) Lowering of spent fuel pool level to 1.5 ft above the fuel racks as indicated by SF-LI-2616 (MPCS computer point A4172) or SF-LI-2617 (MPCS computer point A4220)..

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.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 12 ft. 3 in. above the top of the fuel racks (Level 2) and SFP level 1.5 ft. above the top of the fuel racks (Level 3).

The Spent Fuel Pool Instrumentation System (SFPIS) consists of two new independent level instrument channels (SF-L-2616 and SF-L-2617) in the Spent Fuel system. The SFPIS channels will be used to monitor spent fuel pool level during and following beyond design basis events that could challenge the capability to ensure optimum protection for the stored fuel assemblies in the pool.

Each channel is capable of measuring SFP level over a span from just above the top of the spent fuel racks to the normal SFP operating water level. The SFPIS will be monitored in accordance with Beyond Design Basis guidelines contained in FSGs for Extended Loss of AC Power and Alternate SFP Makeup and Cooling.

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.

RA1

ECL: Alert

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

Operating Mode Applicability: All

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

Notes:

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

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

Monitor	Reading
RM-6528-4 (WRGM rate)	2.85E+6 uCi/sec
RM-6481-1* (MSL A)	10 mR/hr
RM-6482-1* (MSL B)	10 mR/hr
RM-6482-2* (MSL C)	10 mR/hr
RM-6481-2* (MSL D)	10 mR/hr

* With release path to the environment from affected steam line, open ASDV or SRV, line is faulted, or open steam supply to 1-FW-P-37A.

OR

(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary.

OR

(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.

OR

(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.

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.

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: (1 or 2 or 3)

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

OR

(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by high-alarm, or reading in excess of the current high-alarm setpoint **on ANY** of the following radiation monitors:

RM-6518-1	FSB High Range
RM-6562-1	FSB Vent
RM-6535A-1	Manipulator Crane
RM-6535B-1	Manipulator Crane

OR

(3) Lowering of spent fuel pool level to 12 ft. 3 in. above the fuel racks on SF-LI-2616 (MPCS computer point A4172) or SF-LI-2617 (MPCS computer point A4220).

Basis:

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

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.

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 EU1.

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

EAL #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, 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 #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. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event.

EAL #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.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 12 ft. 3 in. above the top of the fuel racks (Level 2) and SFP level 1.5 ft. above the top of the fuel racks (Level 3).

The Spent Fuel Pool Instrumentation System (SFPIS) consists of two new independent level instrument channels (SF-L-2616 and SF-L-2617) in the Spent Fuel system. The SFPIS channels will be used to monitor spent fuel pool level during and following beyond design basis events that could challenge the capability to ensure optimum protection for the stored fuel assemblies in the pool.

Each channel is capable of measuring SFP level over a span from just above the top of the spent fuel racks to the normal SFP operating water level. The SFPIS will be monitored in accordance with Beyond Design Basis guidelines contained in FSGs for Extended Loss of AC Power and Alternate SFP Makeup and Cooling.

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

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2)

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

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

Control Room RM6550
Central Alarm Station (CAS) by survey
Secondary Alarm Station (SAS) by survey

OR

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

Table H1	
Area	Mode
Primary Aux Building 25 ft elevation 7 ft elevation - 26 ft elevation	1, 2, 3, 4
Turbine Building 21 ft elevation 50 ft elevation	1, 2, 3
Essential Switchgear Rooms	1, 2, 3, 4
Waste Process Building 25 ft elevation -3 ft elevation	1, 2, 3
Containment	3, 4
RHR/CBS Equipment Vaults	3, 4

Basis:

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.

IMPEDE: Entry into an area requires extraordinary measures to facilitate entry of personnel into the affected room/area by installing temporary shielding, requiring use of non-routine protective equipment, or requesting an extension in dose limits beyond normal administrative limits.

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 STED/SED should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For EAL #2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area.

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area.
- The action for which room/area entry is required is of an administrative or record keeping nature.
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

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

RU1

ECL: Notification of Unusual Event

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

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2)

Notes:

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

1) a Valid reading on **ANY** of the following effluent monitors greater than 2 times the value of the current high-alarm setpoint for 60 minutes or longer:

RM-6509-1 (WTT Disch)
RM-6521-1 (TB Sump)
RM-6519-1 (SG Blowdown)
RM-6473-1 (WT LIQ EFF)
RM-6528-4 (WRGM rate)

AND

b. The discharge flow to the environment is not isolated within 60 minutes.

OR

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

Basis:

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

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

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

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

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

EAL #1 - This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways. EAL #1 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the ODCM limit and releases are not terminated within 60 minutes. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

The discharge flowpaths associated with RM-6509-1, 6521-1, 6519-1, and 6473-1 have automatic and manual flow isolation capability. The EAL wording addresses a situation where a residual source term exists in a discharge flowpath AFTER the flowpath has been isolated, and the associated radiation monitor remains at values above 2 times the value of the current high-alarm setpoint. EAL 1.b ensures that the Initiating Condition (IC) intent of "to the environment" is met. The 60-minute assessment clock starts at the same time for both EAL 1.a and 1.b (i.e., clocks run concurrently). There must be a release to the environment (i.e., the flowpath cannot be isolated) during the same period that a monitor value is greater than 2 times the value of the current high-alarm setpoint.

EAL #2 - This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways.

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

RU2

ECL: Notification of Unusual Event

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

Operating Mode Applicability: All

Emergency Action Levels:

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

1-SF-LI-2607 (Spent Fuel Pool Level)
1-SF-LI-2629 or 1-SF-LIT-2629-1 (Reactor Refuel Cavity Level)

AND

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

RM-6535-A-1, Containment Manipulator Crane
RM-6535-B-1, Containment Manipulator Crane
RM-6549-1, FSB Spent Fuel Range Low
RM-6518-1, FSB Spent Fuel Range Hi

Basis:

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.

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 or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

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

A drop in water level above irradiated fuel within the reactor vessel 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.

4 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

<u>**Recognition Category "C" Initiating Condition Matrix**</u>

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
CG1 Loss of reactor vessel/RCS inventory affecting fuel clad integrity with containment challenged. *Op. Modes: 5, 6*	**CS1** Loss of reactor vessel/RCS inventory affecting core decay heat removal capability. *Op. Modes: 5, 6*	**CA1** Loss of reactor vessel/RCS inventory. *Op. Modes: 5, 6*	**CU1** UNPLANNED loss of reactor vessel/RCS inventory for 15 minutes or longer. *Op. Modes: 5, 6*
		CA2 Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer. *Op. Modes: 5, 6, Defueled*	**CU2** Loss of all but one AC power source to emergency buses for 15 minutes or longer. *Op. Modes: 5, 6, Defueled*
		CA3 Inability to maintain the plant in cold shutdown. *Op. Modes: 5, 6*	**CU3** UNPLANNED increase in RCS temperature. *Op. Modes: 5, 6*
			CU4 Loss of Vital DC power for 15 minutes or longer. *Op. Modes: 5, 6*
			CU5 Loss of all onsite or offsite communications capabilities. *Op. Modes: 5, 6, Defueled*
		CA6 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. *Op. Modes: 5, 6*	

CG1

ECL: General Emergency

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

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) a. RVLIS Full Range < 55% (-141.5 in) for 30 minutes or longer.

AND

b. **ANY** indication from the Containment Challenge Table C2.

OR

(2) a. Reactor vessel/RCS level cannot be monitored for 30 minutes or longer.

AND

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

RM-6535A-1 (Manipulator Crane) reading greater than 9500 mR/hr
RM-6535B-1 (Manipulator Crane) reading greater than 9500 mR/hr
Erratic source range monitor indication
UNPLANNED increase in Containment Sumps A or B levels of sufficient magnitude to indicate core uncovery.
Visual observation.

AND

c. **ANY** indication from the Containment Challenge Table C2.

Containment Challenge Table C2
CONTAINMENT INTEGRITY not established *
Containment H_2 concentration $\geq$ 6%
UNPLANNED increase in containment pressure

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

Basis:

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 INTEGRITY: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

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 INTEGRITY not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT INTEGRITY 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.

Manipulator Crane setpoint of 9500 mR/hr is 95% of the monitor range.

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

RVLIS LEVEL (%)	VESSEL LEVEL (inches from vessel flange)	
~108	119.8	
100	81.3	
90	31.8	
80	-17.7	
70	-67.2	
63	-101.9	RC-LI-9405, RC-LIT-9467, and the Tygon Tube do not indicate reactor vessel level when actual level is less than -95" due to the weir on the RCP discharge.
60	-116.7	
55	-141.5	
50	-166.2	
40	-215.7	
30	-265.2	
20	-314.7	
10	-364.2	
0	-413.7	

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.*

Reference:
FSAR Table 12.3-14

ECL: Site Area Emergency

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

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2 or 3)

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

(1) a. CONTAINMENT INTEGRITY not established.

AND

b. RVLIS Full Range < 63% (-101.9 in) .

OR

(2) a. CONTAINMENT INTEGRITY established.

AND

b. RVLIS Full Range < 55% (-141.5 in).

OR

(3) a. Reactor vessel/RCS level cannot be monitored for 30 minutes or longer.

AND

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

RM-6535A-1 (Manipulator Crane) reading greater than 9500 mR/hr
RM-6535B-1 (Manipulator Crane) reading greater than 9500 mR/hr
Erratic source range monitor indication
UNPLANNED increase in Containment Sumps A or B levels of sufficient magnitude to indicate core uncovery
Visual observation.

Basis:

CONTAINMENT INTEGRITY: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

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.

This IC addresses a significant and prolonged loss of reactor vessel/RCS 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 INTEGRITY following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs 1.b and 2.b reflect the fact that with CONTAINMENT INTEGRITY established, there is a lower probability of a fission product release to the environment.

In EAL 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. Manipulator Crane setpoint of 9500 mR/hr is 95% of the monitor range.

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

RVLIS LEVEL (%)	VESSEL LEVEL (inches from vessel flange)	
~108	119.8	
100	81.3	
90	31.8	
80	-17.7	
70	-67.2	
63	-101.9	RC-LI-9405, RC-LIT-9467, and the Tygon Tube do not indicate reactor vessel level when actual level is less than -95" due to the weir on the RCP discharge.
60	-116.7	
55	-141.5	
50	-166.2	
40	-215.7	
30	-265.2	
20	-314.7	
10	-364.2	
0	-413.7	

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.

CA1

ECL: Alert

Initiating Condition: Loss of reactor vessel/RCS inventory.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) Loss of reactor vessel/RCS inventory as indicated by RVLIS full range < 64% (-96.9 in.)

OR

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

AND

b. UNPLANNED increase in Containment Sumps A or B levels due to a loss of reactor vessel/RCS inventory.

Basis:

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.

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

For EAL #1, a lowering of water level below 64% indicates that operator actions have not been successful in restoring and maintaining reactor vessel/RCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

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

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

RVLIS LEVEL (%)	VESSEL LEVEL (inches from vessel flange)	
~108	119.8"	
100	81.3"	
90	31.8"	
80	-17.7"	
70	-67.2"	
64	-96.9"	RC-LI-9405, RC-LIT-9467, and the Tygon Tube do not indicate reactor vessel level when actual level is less than -95" due to the weir on the RCP discharge.
60	-116.7"	
50	-166.2"	
40	-215.7"	
30	-265.2"	
20	-314.7"	
10	-364.2"	
0	-413.7"	

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 reactor vessel/RCS inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

CA2

ECL: Alert

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

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Levels:

Note:

- The STED/SED should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
- For a bus to be considered energized from SEPS, both SEPS diesel generator sets must be functional.

(1) Loss of **ALL** offsite and **ALL** onsite AC Power to BOTH AC emergency buses E5 AND E6 for 15 minutes or longer.

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.

1-EDE-SWG-5 (E5) and 1-EDE-SWG-6 (E6) are the 4.16 kV emergency buses for Train A and Train B respectively. These buses supply all safety-related loads.

This Initiating Condition is not met if either Bus E5 or E6 is energized from the Supplemental Emergency Power System (SEPS).

The SEPS primary function is to supply power to one 4.16 kV emergency bus, EDE-SWG-5 (E5) or EDE-SWG-6 (E6), in the event of a loss-of-offsite-power (LOOP) and both EDGs fail to start and load. In addition (SEPS) provides back up power to the emergency buses when one of the emergency diesel generators (EDG) is out of service for up to fourteen days. SEPS can be used when it is anticipated that one of the EDGs will be inoperable for longer than the technical specification allowable outage time (AOT) of 72 hours.

The design of the SEPS is capable of providing the required safety-related loads in the event of a loss of offsite power if both emergency diesel generators fail to start and load. During these events it is assumed that there is no seismic event or an event that requires safeguards actuation (SI, CBS, CVI, CI, etc.). In addition to providing power to the required loads, the total combined output of the SEPS system can supply either the RHR pump or the SI pump and one set of pressurizer heaters. These design conditions are based on Probabilistic Risk Evaluation (PRA) EE-03-007.

The SEPS consists of two 4.16 kV generators which use diesel fuel engines as the prime mover. The generator sets (gensets) SEPS-DG-2-A and SEPS-DG-2-B are capable of automatically starting, synchronizing together and energizing the SEPS electrical bus. The SEPS design requires a "dead bus" transfer back to an offsite power source, i.e., the emergency bus powered by SEPS must be de-energized before restoring offsite power.

For power restoration from the SEPS, both SEPS diesel generator sets must be functional. Both SEPS engines are required to power the emergency bus and equipment required to maintain core cooling.

The use of the SEPS is recognized in the Emergency Operating Procedures.

Reference:
UFSAR Section 8.3.1, AC Power Systems

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: 5, 6

Emergency Action Levels: (1 or 2)

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

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

TableC1 - RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT INTEGRITY Status	Heat-up Duration
INTACT and reactor vessel ≥ -36 inches	Not applicable	60 minutes*
Not INTACT or reactor vessel < -36 inches	Established	20 minutes*
	Not Established	0 minutes
* If RHR is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

OR

(2) UNPLANNED RCS pressure increase greater than 25 psig. (This EAL does not apply during water-solid plant conditions.)

Basis:

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.

INTACT: Capable of being pressurized.

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.

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

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

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

EAL #2 provides a pressure-based indication of RCS heat-up. The wide-range RCS pressure transmitters have a range of 0 to 3,000 psig. The main control boards have two post-accident monitoring qualified meters, one for each wide-range RCS pressure transmitter. These meters have major divisions at 100 psig intervals and minor divisions at 50 psig intervals. Since it is possible to read the approximate mid-point between minor divisions, the value is set to 25 psig.

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: 5, 6

Emergency Action Levels:

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

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

AND

b. **EITHER** of the following:

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

OR

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

Basis:

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.

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.

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.

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

EAL 1.b.1 addresses damage to a SAFETY SYSTEM 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.

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

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

CU1

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of reactor vessel/RCS inventory for 15 minutes or longer.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) UNPLANNED loss of reactor coolant results in reactor vessel/RCS level less than a required lower limit of an operating band, specified by an operating procedure for 15 minutes or longer.

OR

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

AND

b. UNPLANNED increase in Containment Sump A or B level.

Basis:

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.

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

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

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

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

EAL #2 addresses a condition where all means to determine reactor vessel/RCS level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the reactor vessel/RCS.

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

CU2

ECL: Notification of Unusual Event

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

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Levels:

Notes:

- The STED/SED should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
- For power restoration from the SEPS, both SEPS diesel generator sets must be functional.

(1) a. AC power capability to Both AC emergency buses E5 AND E6 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.

NOTE

There are six power sources to consider:

- 345 kV offsite power line 369
- 345 kV offsite power line 363
- 345 kV offsite power line 394
- Emergency Diesel Generator A
- Emergency Diesel Generator B
- SEPS. For SEPS to be considered available, both SEPS diesel generator sets must be functional

Basis:

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.

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.

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

1-EDE-SWG-5 (E5) and 1-EDE-SWG-6 (E6) are the 4.16 kV emergency buses for Train A and Train B respectively. These buses supply all safety-related loads.

This Initiating Condition is not met if either Bus E5 or E6 is energized from the Supplemental Emergency Power System (SEPS).

The SEPS primary function is to supply power to one 4.16 kV emergency bus, EDE-SWG-5 (E5) or EDE-SWG-6 (E6), in the event of a loss-of-offsite-power (LOOP) and both EDGs fail to start and load. In addition (SEPS) provides back up power to the emergency buses when one of the emergency diesel generators (EDG) is out of service for up to fourteen days. SEPS can be used when it is anticipated that one of the EDGs will be inoperable for longer than the technical specification allowable outage time (AOT) of 72 hours.

The design of the SEPS is capable of providing the required safety-related loads in the event of a loss of offsite power if both emergency diesel generators fail to start and load. During these events it is assumed that there is no seismic event or an event that requires safeguards actuation (SI, CBS, CVI, CI, etc.). In addition to providing power to the required loads, the total combined output of the SEPS system can supply either the RHR pump or the SI pump and one set of pressurizer heaters. These design conditions are based on Probabilistic Risk Evaluation (PRA) EE-03-007.

The SEPS consists of two 4.16 kV generators which use diesel fuel engines as the prime mover. The generator sets (gensets) SEPS-DG-2-A and SEPS-DG-2-B are capable of automatically starting, synchronizing together and energizing the SEPS electrical bus. The SEPS design requires a "dead bus" transfer back to an offsite power source, i.e., the emergency bus powered by SEPS must be de-energized before restoring offsite power.

For power restoration from the SEPS, both SEPS diesel generator sets must be functional. Both SEPS engines are required to power the emergency bus and equipment required to maintain core cooling.

The use of the SEPS is recognized in the Emergency Operating Procedures.

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

Reference: UFSAR Section 8.3.1, AC Power Systems

CU3

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED increase in RCS temperature.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

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

OR

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

Basis:

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.

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

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

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

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC 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: 5, 6

Emergency Action Levels:

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

(1) Indicated voltage is less than 105V on required Vital DC buses associated with the Protected Train for 15 minutes or longer.

Train A 11A and 11C
Train B 11B and 11D

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.

Per DBD-ED-05, the DC bus voltage range within which the 125 Volt DC system is considered operable is 105 volts minimum to 140 volts maximum. The vital DC Buses (Switchgear) are SWG-11A and 11C for Train A and SWG-11B and 11D for Train B.

Reference: UFSAR Section 8.3.2, DC Power System
Procedure OS1248.01, Loss of a Vital 125 VDC Bus
Procedure VPRO F5278, Loss of All Vital DC Power
DBD-ED-05, 125 VDC System

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: 5, 6, Defueled

Emergency Action Levels: (1 or 2 or 3)

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

In-Plant (PBX) Telephones
Gai-Tronics
Plant Radio System

OR

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

Nuclear Alert System (NAS)
Backup NAS
Control Room/TSC telephones

OR

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

Emergency Notification System (ENS)
Control Room/TSC telephones
FTS telephones in the TSC

Basis:

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

This IC should be assessed only when extraordinary means are being utilized to make communications possible.

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

EAL #2 addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are Commonwealth of Massachusetts and State of New Hampshire.

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

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

<u>Recognition Category "E" Initiating Condition Matrix</u>

UNUSUAL EVENT

EU1 Damage to a loaded cask CONFINEMENT BOUNDARY.
Op. Modes: All

EU1

ECL: Notification of Unusual Event

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: All

Emergency Action Levels:

Note:

The on-contact dose rate may be determined based on measurement of a dose rate at some distance from the cask

(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by ANY of the following on-contact surface radiation readings greater than:

1600 mrem/hr at the front bird screen

4 mrem/hr at the door centerline

4 mrem/hr at the end shield wall exterior

Basis:

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

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 in the Horizontal Storage Module. 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.

Reference:
Appendix A to Certificate Of Compliance No. 1030 NUHOMS® HD System Generic Technical Specifications 5.4.3.

6 FISSION PRODUCT BARRIER ICS/EALS

Recognition Category "F" Initiating Condition Matrix

GENERAL EMERGENCY	
FG1	Loss of any two barriers and Loss or Potential Loss of the third barrier. *Op. Modes: 1, 3, 2, 4*
SITE AREA EMERGENCY	
FS1	Loss or Potential Loss of any two barriers. *Op. Modes: 1, 3, 2, 4*
ALERT	
FA1	Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier. *Op. Modes: 1, 3, 2, 4*

Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

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

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. RCS or SG Tube Leakage		**1. RCS or SG Tube Leakage**		**1. RCS or SG Tube Leakage**	
Not Applicable	A. Core Cooling (C) CSF – ORANGE entry conditions met (NOTE 1)	A. An automatic or manual SI actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	A. Operation of a second charging pump in the normal charging mode is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS Integrity (P) CSF – RED entry conditions met with RCS press > 300 psig. (NOTE 1).	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable
2. Inadequate Heat Removal		**2. Inadequate Heat Removal**		**2. Inadequate Heat Removal**	
A. Core Cooling (C) CSF – RED entry conditions met. (NOTE 1)	A. Core Cooling (C) CSF – ORANGE entry conditions met. (NOTE1) **OR** B. Heat Sink (H) CSF – RED entry conditions met. (NOTE 1)	Not Applicable	A. Heat Sink (H) CSF – RED entry conditions met. (NOTE 1)	Not Applicable	A. Core Cooling (C) CSF – RED entry conditions met for 15 minutes or longer. (NOTE 1)

3. RCS Activity / Containment Radiation		**3. RCS Activity / Containment Radiation**		**3. RCS Activity / Containment Radiation**	
A. Post LOCA Radiation Monitors RM 6576A-1 or RM 6576B-1 ≥ 95 R/hr. **OR** B. RCS activity > 300 uCi/gm Dose Equivalent I 131 as determined per Procedure CS0925.01, Reactor Coolant Post Accident Sampling.	Not Applicable	A. Post LOCA Radiation Monitors RM 6576A-1 or RM 6576B-1 ≥ 16 R/hr.	Not Applicable	Not Applicable	A. Post LOCA Radiation Monitors RM 6576A-1 or RM 6576B-1 ≥ 1,305 R/hr. .
4. Containment Integrity or Bypass		**4. Containment Integrity or Bypass**		**4. Containment Integrity or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Containment isolation is required **AND** **EITHER** of the following: 1. Containment integrity has been lost based on STED/SED judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment (Z) CSF – RED entry conditions met. (NOTE 1) **OR** B. Containment H_2 concentration ≥ 6% **OR** C. 1. Containment pressure > 18 psig **AND** 2. Less than one full train of Containment Building Spray (CBS) is operating per design for 15 minutes or longer.

5. **STED/SED Judgment**		5. **STED/SED Judgment**		5. **STED/SED Judgment**	
A. **ANY** condition in the opinion of the STED/SED that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the STED/SED that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the STED/SED that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the STED/SED that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the STED/SED that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the STED/SED that indicates Potential Loss of the Containment Barrier.

NOTE 1: Refer to ER 1.1, Section 1.1, Discussion concerning the proper use of CSFSTs as EALs

Basis Information For Fission Product Barrier Table

FUEL CLAD BARRIER THRESHOLDS:

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

1. RCS or SG Tube Leakage

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

Potential Loss 1.A

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

2. Inadequate Heat Removal

Loss 2.A

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

Potential Loss 2.A

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

Potential Loss 2.B

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

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

As a potential loss indication, developers should consider including a threshold the same as, or similar to, "Core Cooling Orange entry conditions met" in accordance with the guidance at the front of this section.

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

3. RCS Activity / Containment Radiation

Loss 3.A

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

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

<u>Loss 3.B</u>

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

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

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

5. STED/SED Judgment

<u>Loss 5.A</u>

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

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

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

RCS BARRIER THRESHOLDS:

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

1. RCS or SG Tube Leakage

Loss 1.A

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

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

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

Potential Loss 1.A

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

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

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

Potential Loss 1.B

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

2. Inadequate Heat Removal

There is no Loss threshold associated with Inadequate Heat Removal.

Potential Loss 2.A

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

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

3. **RCS Activity / Containment Radiation**

Loss 3.A

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

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

4. **Containment Integrity or Bypass**

Not Applicable (included for numbering consistency)

5. **STED/SED Judgment**

Loss 5.A

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

Potential Loss 5.A

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

CONTAINMENT BARRIER THRESHOLDS:

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

1. RCS or SG Tube Leakage

Loss 1.A

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

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably [*part of the FAULTED definition*] and the faulted steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC MU3 for the fuel clad barrier (i.e., RCS activity values) and IC MU4 for the RCS barrier (i.e., RCS leak rate values).

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

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component. These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

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

P-to-S Leak Rate	Affected SG is FAULTED Outside of Containment? Yes	No
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per MU4	Unusual Event per MU4
Requires operation of a second charging pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1	Alert per FA1
Requires an automatic or manual SI actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1	Alert per FA1

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

2. Inadequate Heat Removal

There is no Loss threshold associated with Inadequate Heat Removal.

Potential Loss 2.A

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

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

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

3. RCS Activity / Containment Radiation

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

Potential Loss 3.A

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

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

4. **Containment Integrity or Bypass**

Loss 4.A

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

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

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

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

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

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

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

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

Loss 4.B

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

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

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

Potential Loss 4.A

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

Potential Loss 4.B

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

Potential Loss 4.C

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems are either lost or performing in a degraded manner.

5. STED/SED Judgment

Loss 5.A

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

Potential Loss 5.A

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

7 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

Recognition Category "H" Initiating Condition Matrix

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
	HS1 HOSTILE ACTION within the PROTECTED AREA. *Op. Modes: All*	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. *Op. Modes: All*	**HU1** Confirmed SECURITY CONDITION or threat. *Op. Modes: All*
			HU2 Seismic event greater than OBE levels. *Op. Modes: All*
			HU3 Hazardous event. *Op. Modes: All*
			HU4 FIRE potentially degrading the level of safety of the plant. *Op. Modes: All*
		HA5 Gaseous release impeding access to equipment necessary for normal plant operations, shutdown or cooldown. *Op. Modes: All*	
	HS6 Inability to control a key safety function from outside the Control Room. *Op. Modes: All*	**HA6** Control Room evacuation resulting in transfer of plant control to alternate locations. *Op. Modes: All*	
HG7 Other conditions exist which in the judgment of the STED/SED warrant declaration of a General Emergency. *Op. Modes: All*	**HS7** Other conditions exist which in the judgment of the STED/SED warrant declaration of a Site Area Emergency. *Op. Modes: All*	**HA7** Other conditions exist which in the judgment of the STED/SED warrant declaration of an Alert. *Op. Modes: All*	**HU7** Other conditions exist which in the judgment of the STED/SED warrant declaration of an Unusual Event. *Op. Modes: All*

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ECL: General Emergency

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

Operating Mode Applicability: All

Emergency Action Levels:

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

Basis:

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

HS1

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Levels:

Note: This Initiating Condition and EAL do not apply to an attack solely on the Dry Fuel Storage Protected Area. An attack on the Dry Fuel Storage Facility Protected Area should be considered an attack within the Owner Controlled Area and classified as an Alert per Initiating Condition HA1.

(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by security shift supervision

Basis:

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.

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

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

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

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

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

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

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

Escalation of the emergency classification level would be via IC HG7.

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 STED/SED should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

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

AND

b. Control of **ANY** of the following key safety functions is not reestablished within 15 minutes.

Reactivity control
Core cooling
RCS heat removal

Basis:

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

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on STED/SED judgment. The STED/SED is expected to make a reasonable, informed judgment within 15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

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

ECL: Site Area Emergency

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

Operating Mode Applicability: All

Emergency Action Levels:

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

Basis:

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.

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

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: (1 or 2)

(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA or the Dry Fuel Storage Facility as reported by security shift supervision.

OR

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

Basis:

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.

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

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

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

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

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures. The Alert declaration will also heighten the awareness of Offsite Response Organizations, allowing them to be better prepared should it be necessary to consider further actions.

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

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

EAL #2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with site procedures.

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.

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

ECL: Alert

Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, shutdown or cooldown.

Operating Mode Applicability: All

Emergency Action Levels:

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

(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into any Table H1 rooms or areas.

AND

b. Entry into the room or area is prohibited or IMPEDED.

Table H1	
Area	Mode
Primary Aux Building 25 ft elevation 7 ft elevation - 26 ft elevation	1, 2, 3, 4
Turbine Building 21 ft elevation 50 ft elevation	1, 2, 3
Essential Switchgear Rooms	1, 2, 3, 4
Waste Process Building 25 ft elevation -3 ft elevation	1, 2, 3
Containment	3, 4
RHR/CBS Equipment Vaults	3, 4

Basis:

IMPEDE: Entry into an area requires extraordinary measures to facilitate entry of personnel into the affected room/area by installing temporary shielding, requiring use of non-routine protective equipment, or requesting an extension in dose limits beyond normal administrative limits.

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the STED/SED's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area.

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area.
- The action for which room/area entry is required is of an administrative or record keeping nature.
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

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

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

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

References:
OS1000.03, Plant Shutdown From Minimum Load to Hot Standby
OS1000.04, Plant Cooldown From Hot Standby to Cold Shutdown

HA6

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels:

(1) Entry into Procedure OS1200.02 for control room evacuation resulted in plant control being transferred from the Control Room to Remote Safe Shutdown components.

Basis:

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

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

Escalation of the emergency classification level would be via IC HS6.

HA7

ECL: Alert

Initiating Condition: Other conditions exist which in the judgment of the STED/SED warrant declaration of an Alert.

Operating Mode Applicability: All

Emergency Action Levels:

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

Basis:

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.

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

ECL: Notification of Unusual Event

Initiating Condition: Confirmed SECURITY CONDITION or threat.

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2 or 3)

(1) A Code Yellow is reported by the Security Shift Supervisor.

OR

(2) Notification of a credible security threat directed at Seabrook Station.

OR

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

Basis:

Code Yellow - 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 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.

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

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

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

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

EAL #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with site procedures.

EAL #3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with site procedures.

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

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

References:
OS1290.03, Response to a Security Event.
OS1290.04, Response to an Airborne Security Event

HU2

ECL: Notification of Unusual Event

Initiating Condition: Seismic event greater than OBE levels.

Operating Mode Applicability: All

Emergency Action Levels:

(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by:

a. The red "EVENT" light is lit on seismic monitoring control panel 1-SM-CP-58.

AND

b. The yellow "OBE" light is lit on seismic monitoring control panel 1-SM-CP-58.

OR

(2) a. Seismic monitoring system out of service

AND

b. Control Room personnel feel an actual or potential seismic event

AND

c. The occurrence of a seismic event is confirmed in a manner deemed appropriate by the Shift Manager

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 Safe Shutdown Earthquake (SSE) 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. 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 STED/SED may seek external verification if deemed appropriate; however, the verification action must not preclude a timely emergency declaration.

Reference:
EC 282184, Seismic Monitoring System Upgrade

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or MA9.

HU3

ECL: Notification of Unusual Event

Initiating Condition: Hazardous event.

Operating Mode Applicability: All

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

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

(1) A tornado strike within the PROTECTED AREA.

OR

(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.

OR

(3) Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials.

OR

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

Basis:

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.

IMPEDE: Entry into an area requires extraordinary measures to facilitate entry of personnel into the affected room/area by installing temporary shielding, requiring use of non-routine protective equipment, or requesting an extension in dose limits beyond normal administrative limits.

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

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

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

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

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

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

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

HU4

ECL: Notification of Unusual Event

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

Operating Mode Applicability: All

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

Notes:

- The STED/SED should declare the Unusual Event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- A containment fire alarm is considered valid upon receipt of multiple zones (more than 1) actuated on CP-376 panel.

(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 H2 plant rooms or areas:

Table H2	
Condensate Storage Tank Enclosure	Fuel Storage Building
Containment	Primary Auxiliary Building
Control Building	Service Water Pump House
Cooling Tower	Steam and Feedwater Pipe Chases
Diesel Generator Building	North Tank Farm
Emergency Feedwater Pump House	Startup Feedwater Pump Area
RHR/CBS Equipment Vaults	

OR

(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 Table H2 plant rooms or areas except Containment in Modes 1 and 2 (see note above):

AND

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

OR

(3) A FIRE within the plant PROTECTED AREA or Dry Fuel Storage Facility not extinguished within 60-minutes of the initial report, alarm or indication.

OR

(4) A FIRE within the plant PROTECTED AREA or Dry Fuel Storage Facility that requires firefighting support by an offsite fire response agency to extinguish.

Basis:

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.

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.

With regard to containment fire alarms, there is constant air movement in containment due to the operation of the CAH system. The operating cooling units are drawing air to the units past the smoke detectors. It can reasonably be expected that a fire that burns for 15 minutes would produce sufficient products of combustion to cause fire detectors in multiple zones to alarm.

<u>EAL #1</u>

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished. 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 #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 #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.

<u>EAL #3</u>

In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA or the Dry Fuel Storage Facility.

EAL #4

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

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or MA9.

HU7

ECL: Notification of Unusual Event

Initiating Condition: Other conditions exist which in the judgment of the STED/SED warrant declaration of an Unusual Event.

Operating Mode Applicability: All

Emergency Action Levels:

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

Basis:

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

8 SYSTEM MALFUNCTION ICS/EALS

Recognition Category "M" Initiating Condition Matrix

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
MG1 Prolonged loss of all offsite and all onsite AC power to emergency buses. *Op. Modes: 1, 2, 3, 4*	**MS1** Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer. *Op. Modes: 1, 2, 3, 4*	**MA1** Loss of all but one AC power source to emergency buses for 15 minutes or longer. *Op. Modes: 1, 2, 3, 4*	**MU1** Loss of all offsite AC power capability to emergency buses for 15 minutes or longer. *Op. Modes: 1, 2, 3, 4*
		MA2 UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. *Op. Modes: 1, 2, 3, 4*	**MU2** UNPLANNED loss of Control Room indications for 15 minutes or longer. *Op. Modes: 1, 2, 3, 4*
			MU3 Reactor coolant activity greater than Technical Specification allowable limits. *Op. Modes: 1, 2, 3, 4*
			MU4 RCS leakage for 15 minutes or longer. *Op. Modes: 1, 2, 3, 4*
	MS5 Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal. *Op. Modes: 1*	**MA5** Automatic or manual trip fails to shutdown the reactor and subsequent manual actions taken at the Main Control Board are not successful in shutting down the reactor. *Op. Modes: 1*	**MU5** Automatic or manual trip fails to shutdown the reactor . *Op. Modes: 1*
			MU6 Loss of all onsite or offsite communications capabilities. *Op. Modes: 1, 2, 3, 4*
			MU7 Failure to isolate containment or loss of containment pressure control. *Op. Modes: 1, 2, 3, 4*
MG8 Loss of all AC and Vital DC power sources for 15 minutes or longer. *Op. Modes: 1, 2, 3, 4*	**MS8** Loss of all Vital DC power for 15 minutes or longer. *Op. Modes: 1, 2, 3, 4*		
		MA9 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. *Op. Modes: 1, 2, 3, 4*	

ECL: General Emergency

Initiating Condition: Prolonged loss of all offsite and all onsite AC power to emergency buses.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Notes:

- The STED/SED should declare the General Emergency promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.
- For a bus to be considered energized from SEPS, both SEPS diesel generator sets must be functional.

(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to BOTH AC emergency buses E5 AND E6.

AND

b. **ANY** of the following:

Restoration of at least one AC emergency bus in less than 4 hours is not likely.
Core Cooling (C) CSF RED entry conditions met

Basis:

This IC addresses a prolonged loss of all power sources to AC emergency 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 emergency bus by the end of the analyzed 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 emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, 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.

This Initiating Condition is not met if either Bus E5 or E6 is energized from the Supplemental Emergency Power System (SEPS).

The SEPS primary function is to supply power to one 4.16 kV emergency bus, EDE-SWG-5 (E5) or EDE-SWG-6 (E6), in the event of a loss-of-offsite-power (LOOP) and both EDGs fail to start and load. In addition (SEPS) provides back up power to the emergency buses when one of the emergency diesel generators (EDG) is out of service for up to fourteen days. SEPS can be used when it is anticipated that one of the EDGs will be inoperable for longer than the technical specification allowable outage time (AOT) of 72 hours.

The design of the SEPS is capable of providing the required safety-related loads in the event of a loss of offsite power if both emergency diesel generators fail to start and load. During these events it is assumed that there is no seismic event or an event that requires safeguards actuation (SI, CBS, CVI, CI, etc.). In addition to providing power to the required loads, the total combined output of the SEPS system can supply either the RHR pump or the SI pump and one set of pressurizer heaters. These design conditions are based on Probabilistic Risk Evaluation (PRA) EE-03-007.

The SEPS consists of two 4.16 kV generators which use diesel fuel engines as the prime mover. The generator sets (gensets) SEPS-DG-2-A and SEPS-DG-2-B are capable of automatically starting, synchronizing together and energizing the SEPS electrical bus. The SEPS design requires a "dead bus" transfer back to an offsite power source, i.e., the emergency bus powered by SEPS must be de-energized before restoring offsite power.

For power restoration from the SEPS, both SEPS diesel generator sets must be functional. Both SEPS engines are required to power the emergency bus and equipment required to maintain core cooling.

The use of the SEPS is recognized in the Emergency Operating Procedures

Reference: UFSAR Section 8.3.1, AC Power Systems

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, 4

Emergency Action Levels:

Note:

- The STED/SED should declare the General Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
- For a bus to be considered energized from SEPS, both SEPS diesel generator sets must be functional.

(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to BOTH AC emergency buses E5 AND E6 for 15 minutes or longer.

AND

b. Indicated voltage is less than 105 V on ALL Vital DC buses 11A, 11B, 11C and 11D for 15 minutes or longer.

Basis:

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

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

This Initiating Condition is not met if either Bus E5 or E6 is energized from the Supplemental Emergency Power System (SEPS).

The SEPS primary function is to supply power to one 4.16 kV emergency bus, EDE-SWG-5 (E5) or EDE-SWG-6 (E6), in the event of a loss-of-offsite-power (LOOP) and both EDGs fail to start and load. In addition (SEPS) provides back up power to the emergency buses when one of the emergency diesel generators (EDG) is out of service for up to fourteen days. SEPS can be used when it is anticipated that one of the EDGs will be inoperable for longer than the technical specification allowable outage time (AOT) of 72 hours.

The design of the SEPS is capable of providing the required safety-related loads in the event of a loss of offsite power if both emergency diesel generators fail to start and load. During these events it is assumed that there is no seismic event or an event that requires safeguards actuation (SI, CBS, CVI, CI, etc.). In addition to providing power to the required loads, the total combined output of the SEPS system can supply either the RHR pump or the SI pump and one set of pressurizer heaters. These design conditions are based on Probabilistic Risk Evaluation (PRA) EE-03-007.

The SEPS consists of two 4.16 kV generators which use diesel fuel engines as the prime mover. The generator sets (gensets) SEPS-DG-2-A and SEPS-DG-2-B are capable of automatically starting, synchronizing together and energizing the SEPS electrical bus. The SEPS design requires a "dead bus" transfer back to an offsite power source, i.e., the emergency bus powered by SEPS must be de-energized before restoring offsite power.

For power restoration from the SEPS, both SEPS diesel generator sets must be functional. Both SEPS engines are required to power the emergency bus and equipment required to maintain core cooling.

The use of the SEPS is recognized in the Emergency Operating Procedures

Reference: UFSAR Section 8.3.1, AC Power Systems

ECL: Site Area Emergency

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note:

- The STED/SED should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
- For a bus to be considered energized from SEPS, both SEPS diesel generator sets must be functional.

(1) Loss of **ALL** offsite and **ALL** onsite AC power to BOTH AC emergency buses E5 AND E6 for 15 minutes or longer.

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 MG1.

This Initiating Condition is not met if either Bus E5 or E6 is energized from the Supplemental Emergency Power System (SEPS).

The SEPS primary function is to supply power to one 4.16 kV emergency bus, EDE-SWG-5 (E5) or EDE-SWG-6 (E6), in the event of a loss-of-offsite-power (LOOP) and both EDGs fail to start and load. In addition (SEPS) provides back up power to the emergency buses when one of the emergency diesel generators (EDG) is out of service for up to fourteen days. SEPS can be used when it is anticipated that one of the EDGs will be inoperable for longer than the technical specification allowable outage time (AOT) of 72 hours.

The design of the SEPS is capable of providing the required safety-related loads in the event of a loss of offsite power if both emergency diesel generators fail to start and load. During these events it is assumed that there is no seismic event or an event that requires safeguards actuation (SI, CBS, CVI, CI, etc.). In addition to providing power to the required loads, the total combined output of the SEPS system can supply either the RHR pump or the SI pump and one set of pressurizer heaters. These design conditions are based on Probabilistic Risk Evaluation (PRA) EE-03-007.

The SEPS consists of two 4.16 kV generators which use diesel fuel engines as the prime mover. The generator sets (gensets) SEPS-DG-2-A and SEPS-DG-2-B are capable of automatically starting, synchronizing together and energizing the SEPS electrical bus. The SEPS design requires a "dead bus" transfer back to an offsite power source, i.e., the emergency bus powered by SEPS must be de-energized before restoring offsite power.

For power restoration from the SEPS, both SEPS diesel generator sets must be functional. Both SEPS engines are required to power the emergency bus and equipment required to maintain core cooling.

The use of the SEPS is recognized in the Emergency Operating Procedures

Reference: UFSAR Section 8.3.1, AC Power Systems

ECL: Site Area Emergency

Initiating Condition: Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal.

Operating Mode Applicability: 1

Emergency Action Levels:

(1) a. An automatic or manual trip did not shutdown the reactor.

AND

b. All manual actions to shutdown the reactor have been unsuccessful.

AND

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

Core Cooling (C) CSF RED entry conditions met.
Heat Sink (H) CSF RED entry conditions met.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip 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.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

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

ECL: Site Area Emergency

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note:	The STED/SED should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Indicated voltage is less than 105V on **ALL** vital DC buses 11A, 11B, 11C and 11D buses for 15 minutes or longer.

Basis:

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

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or MG8.

ECL: Alert

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Notes:

- The STED/SED should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
- For a bus to be considered energized from SEPS, both SEPS diesel generator sets must be functional.

1) a AC power capability to BOTH AC emergency buses E5 AND E6 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.

NOTE

There are six power sources to consider:

- 345 kV offsite power line 369
- 345 kV offsite power line 363
- 345 kV offsite power line 394
- Emergency Diesel Generator A
- Emergency Diesel Generator B
- SEPS. For SEPS to be considered available, both SEPS diesel generator sets must be functional

Basis:

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.

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 MU1.

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.

- A loss of all offsite power and loss of all emergency power sources with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources with a single train of emergency buses being back-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 MS1.

This Initiating Condition is not met if either Bus E5 or E6 is energized from the Supplemental Emergency Power System (SEPS).

The SEPS primary function is to supply power to one 4.16 kV emergency bus, EDE-SWG-5 (E5) or EDE-SWG-6 (E6), in the event of a loss-of-offsite-power (LOOP) and both EDGs fail to start and load. In addition (SEPS) provides back up power to the emergency buses when one of the emergency diesel generators (EDG) is out of service for up to fourteen days. SEPS can be used when it is anticipated that one of the EDGs will be inoperable for longer than the technical specification allowable outage time (AOT) of 72 hours.

The design of the SEPS is capable of providing the required safety-related loads in the event of a loss of offsite power if both emergency diesel generators fail to start and load. During these events it is assumed that there is no seismic event or an event that requires safeguards actuation (SI, CBS, CVI, CI, etc.). In addition to providing power to the required loads, the total combined output of the SEPS system can supply either the RHR pump or the SI pump and one set of pressurizer heaters. These design conditions are based on Probabilistic Risk Evaluation (PRA) EE-03-007.

The SEPS consists of two 4.16 kV generators which use diesel fuel engines as the prime mover. The generator sets (gensets) SEPS-DG-2-A and SEPS-DG-2-B are capable of automatically starting, synchronizing together and energizing the SEPS electrical bus. The SEPS design requires a "dead bus" transfer back to an offsite power source, i.e., the emergency bus powered by SEPS must be de-energized before restoring offsite power.

For power restoration from the SEPS, both SEPS diesel generator sets must be functional. Both SEPS engines are required to power the emergency bus and equipment required to maintain core cooling.

The use of the SEPS is recognized in the Emergency Operating Procedures

Reference: UFSAR Section 8.3.1, AC Power Systems

MA2

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, 4

Emergency Action Levels:

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

(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
RCS Level
RCS Pressure
Core Exit Temperature
Level in at least two steam generators
Steam Generator Emergency Feed Water Flow

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 trip
SI actuation

Basis:

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.

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, core cooling 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 reactor vessel 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 trip fails to shutdown the reactor, and subsequent manual actions taken at the Main Control Board are not successful in shutting down the reactor.

Operating Mode Applicability: 1

Emergency Action Level:

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.

(1) a. An automatic or manual trip did not shutdown the reactor.
AND
b. Manual actions taken at the MCB are not successful in shutting down the reactor.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the MCB 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 MCB since this event entails a significant failure of the RPS.

A manual action at the MCB is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core. 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 MCB. 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 MCB".

The plant response to the failure of an automatic or manual reactor trip 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 core cooling or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC MS5. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC MS5 or FS1, an Alert declaration is appropriate for this event.

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

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

ECL: Alert

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

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

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

AND

b. **EITHER** of the following:

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

OR

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

Basis:

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.

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.

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.

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

EAL 1.b.1 addresses damage to a SAFETY SYSTEM 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.

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

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

MU1

ECL: Notification of Unusual Event

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

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

(1) Loss of **ALL** offsite AC power capability to BOTH AC emergency buses E5 AND E6 for 15 minutes or longer.

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 emergency buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency 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 MA1.

ECL: Notification of Unusual Event

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

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

(1) 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
RCS Level
RCS Pressure
Core Exit Temperature
Level in at least two steam generators
Steam Generator Emergency Feed Water Flow

Basis:

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.

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, core cooling 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 reactor vessel 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 MA2.

MU3

ECL: Notification of Unusual Event

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2)

(1) RM-6520-1reading greater than 2,670 mR/hr.

OR

(2) Sample analysis indicates that a reactor coolant activity value is greater than the Limiting Condition for Operation (LCO) specified in Technical Specification 3.4.8 Reactor Coolant System Specific Activity.

Basis:

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

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

ECL: Notification of Unusual Event

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2 or 3)

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

(1) RCS unidentified or PRESSURE BOUNDARY LEAKAGE greater than 10 gpm for 15 minutes or longer.

OR

(2) RCS IDENTIFIED LEAKAGE greater than 25 gpm for 15 minutes or longer.

OR

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

Basis:

IDENTIFIED LEAKAGE

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary to secondary leakage).

PRESSURE BOUNDARY LEAKAGE

a. PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary to secondary leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

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

EAL #1 and EAL #2 are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). EAL #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). EAL #1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. For PWRs, an emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected.

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.

MU5

ECL: Notification of Unusual Event

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

Operating Mode Applicability: 1

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 Levels: (1 or 2)

(1) a. An automatic trip did not shutdown the reactor.

AND

b. A subsequent manual action taken at the MCB is successful in shutting down the reactor.

OR

(2) a. A manual trip did not shutdown the reactor.

AND

b. **EITHER** of the following:

1. A subsequent manual action taken at the MCB is successful in shutting down the reactor.

OR

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

Basis:

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

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the MCB to shutdown the reactor. If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the MCB to shutdown the reactor. Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip 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 MCB is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core. 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 MCB".

The plant response to the failure of an automatic or manual reactor trip 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 MCB are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC MA5. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC MA5 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work, the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip 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 trip failure is determined through other means, then this IC and the EALs are not applicable and no classification is warranted.

MU6

ECL: Notification of Unusual Event

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2 or 3)

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

In-Plant (PBX) Telephones
Gai-Tronics
Plant Radio System

OR

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

Nuclear Alert System (NAS)
Backup NAS
Control Room/TSC telephones

OR

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

Emergency Notification System (ENS)
Control Room/TSC telephones
FTS telephones in the TSC

Basis:

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

This IC should be assessed only when extraordinary means are being utilized to make communications possible.

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

EAL #2 addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the Commonwealth of Massachusetts and State of New Hampshire.

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

ECL: Notification of Unusual Event

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2)

(1) a. Failure of containment to isolate when required by an actuation signal.

AND

b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal.

OR

(2) a. Containment pressure greater than 18 psig.

AND

b. Less than one full train of Containment Building Spray (CBS) is operating per design for 15 minutes or longer.

Basis:

This IC addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For EAL #1, the containment isolation signal must be generated as the result on an off-normal/accident condition; a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

EAL #2 addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC	Alternating Current
AOP	Abnormal Operating Procedure
ATWS	Anticipated Transient Without Scram
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
CTMT/CNMT	Containment
CSF	Critical Safety Function
CSFST	Critical Safety Function Status Tree
DBA	Design Basis Accident
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
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GE	General Emergency
IC	Initiating Condition
ID	Inside Diameter
ISFSI	Independent Spent Fuel Storage Installation (Dry Fuel Storage Facility)
Keff	Effective Neutron Multiplication Factor
LCO	Limiting Condition of Operation
LOCA	Loss of Coolant Accident
MCB	Main Control Board
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
mR, mRem, mrem, mREM	milli-Roentgen Equivalent Man
MW	Megawatt
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NORAD	North American Aerospace Defense Command
OBE	Operating Basis Earthquake
OCA	Owner Controlled Area
ODCM	Offsite Dose Calculation Manual
ORO	Off-site Response Organization
PA	Protected Area
PAG	Protective Action Guideline
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
PSIG	Pounds per Square Inch Gauge
R	Roentgen
RCS	Reactor Coolant System
Rem, rem, REM	Roentgen Equivalent Man
RPS	Reactor Protection System

RPV .... Reactor Pressure Vessel
RVLIS .... Reactor Vessel Level Instrumentation System
SAR .... Safety Analysis Report
SAS .... Secondary Alarm Station
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SG .... Steam Generator
SI .... Safety Injection
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
WOG .... Westinghouse Owners Group

APPENDIX 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 NEI 99-01 emergency classification scheme.

Emergency Action Level (EAL): A pre-determined, 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, 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.

CONTAINMENT INTEGRITY:– The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

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

FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

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

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

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

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

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

IMPEDE: Entry into an area requires extraordinary measures to facilitate entry of personnel into the affected room/area by installing temporary shielding, requiring use of non-routine protective equipment, or requesting an extension in dose limits beyond normal administrative limits.

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. (Dry Fuel Storage Facility)

INTACT: Capable of being pressurized.

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

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

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

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

RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

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.

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

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

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

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Enclosure 4 to SBK-L-16196

Enclosure 4 NEI 99-01, Rev. 6, Deviations and Differences, Seabrook Station Nuclear Power Plant – Unit 1

NEI 99-01 Rev 6

Deviations and Differences

Seabrook Station Nuclear Power Plant – Unit 1

GENERIC DIFFERENCES	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
References BWRs	Deleted BWR references as appropriate
Uses A for radiological effluent/radiation level ICs	Uses R for radiological effluent/radiation level ICs
Uses E-HU for ISFSI ICs	Uses EU for ISFSI ICs
Uses S for System Malfunction ICs	Uses M for System Malfunction ICs
Emergency Classification ICs are presented in ascending order (NOUE – GE)	Emergency Classification ICs are presented in descending order (GE – NOUE)

GENERAL NOTES
All NOTEs made site specific by identifying the STED/SED as the user.
Site specific information is highlighted in yellow.

No deviations were indicated to complete the upgrade to Revision 6.

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

RG1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:
(site-specific monitor list and threshold values)

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

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

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

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

Monitor	**Reading**	
RM-6528-4 (WRGM rate)	2.85E+8 uCi/sec	
	Time After Shutdown Reading	
	≤ 1 hr	**> 1 hr to ≤ 2 hrs**
RM-6481-1* (MSL A)	1310 mR/hr	1060 mR/hr
RM-6482-1* (MSL B)	1310 mR/hr	1060 mR/hr
RM-6482-2* (MSL C)	1310 mR/hr	1060 mR/hr
RM-6481-2* (MSL D)	1310 mR/hr	1060 mR/hr

* With release path to the environment from affected steam line, e.g., open ASDV or SRV, line is faulted, open steam supply to 1 FW P 37A.

(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.

(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 /Justification

RG1.1: Site specific information, see V3 EPCALC-06-02 - Effluent Monitor Values for R EALs
RG1.2 & 3: Site specific information, see V4 ODCM and TS Basis for Site Boundary Receptor Point

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

RG2: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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 1.5 ft. (Level3) for 60 minutes or longer.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(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 1.5 ft. above the fuel racks for 60 minutes or longer as indicated by SF-LI-2616 (MPCS computer point A4172) or SF-LI-2617 (MPCS computer point A4220)
Difference /Justification	
RG2.1: Site specific information, see V2 SFP Levels Drawing	

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

RS1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

<table>
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(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)

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

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

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

</td>
<td>

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

Monitor	**Reading**	
RM 6528 4 (WRGM rate)	2.85E+7 uCi/sec	
	Time After Shutdown Reading	
	≤ 1 hr	**> 1 hr to ≤ 2 hrs**
RM-6481-1* (MSL A)	130 mR/hr	100 mR/hr
RM-6482-1* (MSL B)	130 mR/hr	100 mR/hr
RM-6482-2* (MSL C)	130 mR/hr	100 mR/hr
RM-6481-2* (MSL D)	130 mR/hr	100 mR/hr

* With release path to the environment from affected steam line, e.g., open ASDV or SRV, line is faulted, open steam supply to 1-FW-P-37A, etc.

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

(3) 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.

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Difference /Justification

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RS1.1: Site specific information, see V3 EPCALC-06-02 - Effluent Monitor Values for R EALs
RS1.2 & 3: Site specific information, see V4 ODCM and TS Basis for Site Boundary Receptor Point

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</tr>
</table>

RS2: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Spent fuel pool level at (site-specific Level 3 description).	Spent fuel pool level at 1.5 ft.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) Lowering of spent fuel pool level to (site-specific Level 3 value).	(1) Lowering of spent fuel pool level to 1.5 ft above the fuel racks as indicated by SF-LI-2616 (MPCS computer point A4172) or SF-LI-2617 (MPCS computer point A4220).
Difference /Justification	
RS2.1: Site specific information, see V2 SFP Levels Drawing	

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

RA1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

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(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:
(site-specific monitor list and threshold values)
(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).
(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.
(4) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point):
• Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
• Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

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<td>

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

<table>
<tr><th>Monitor</th><th>Readings</th></tr>
<tr><td>RM-6528-4 (WRGM rate)</td><td>2.85E+6 uCi/sec</td></tr>
<tr><td>RM-6481-1* (MSL A)</td><td>10 mR/hr</td></tr>
<tr><td>RM-6482-1* (MSL B)</td><td>10 mR/hr</td></tr>
<tr><td>RM-6482-2* (MSL C)</td><td>10 mR/hr</td></tr>
<tr><td>RM-6481-2* (MSL D)</td><td>10 mR/hr</td></tr>
</table>

* With release path to the environment from affected steam line, e.g., open ASDV or SRV, line is faulted, open steam supply to 1-FW-P-37A, etc.

(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary.
(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.
(4) Field survey results indicate EITHER of the following at or beyond the site boundary:

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

</td>
</tr>
<tr><td colspan="2">Difference /Justification</td></tr>
<tr><td colspan="2">RA1.1: Site specific information, see V3 EPCALC-06-02 - Effluent Monitor Values for R EALs
RA1.2, 3, 4: Site specific information, see V4 ODCM and TS Basis for Site Boundary Receptor Point</td></tr>
</table>

RA2: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Significant lowering of water level above, or damage to, irradiated fuel.	Significant lowering of water level above, or damage to, irradiated fuel.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) Uncovery of irradiated fuel in the REFUELING PATHWAY. (2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the following radiation monitors: (site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms) (3) Lowering of spent fuel pool level to (site-specific Level 2 value). [*See Developer Notes*]	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY. (2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by high-alarm, or reading in excess of the current high-alarm setpoint on **ANY** of the following radiation monitors: RM-6518-1, FSB High Range RM-6562-1, FSB Vent RM-6535A-1, Manip Crane RM-6535B-1, Manip Crane (3) Lowering of spent fuel pool level to 12 ft. 3 inches above the fuel racks on SF-LI-2616 (MPCS computer point A4172) or SF-LI-2617 (MPCS computer point A4220).
Difference /Justification	
RA2.1: Site specific information, see V6 Refueling pathway RU2 RA2 **RA2.2:** Site specific information, see V5 UFSAR Table 12.3-14 - CTMT Post-LOCA Range **RA2.3:** Site specific information, see V2 SFP Levels Drawing	

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

RA3: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.	Radiation levels that impede access to equipment necessary for normal plant operations, shutdown or cooldown.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

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(1) Dose rate greater than 15 mR/hr in **ANY** of the following areas:
- Control Room
- Central Alarm Station
- (other site-specific areas/rooms)

(2) An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified)

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

Control Room
Central Alarm Station (CAS) by survey
Secondary Alarm Station (SAS) by survey

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

Table H1

Area	Mode
Primary Aux Building 25 ft elevation 7 ft elevation - 26 ft elevation	1, 2, 3, 4
Turbine Building 21 ft elevation 50 ft elevation	1, 2, 3
Essential Switchgear Rooms ~~Essential~~ ~~Non-essential~~	1, 2, 3, 4
~~Steam and Feedwater Pipe chases~~	~~1, 2, 3~~
Waste Process Building 25 ft elevation -3 ft elevation ~~-31 ft elevation~~	1, 2, 3
Containment	3, 4
RHR/CBS Equipment Vaults	3, 4

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</tr>
</table>

Difference /Justification

Table H1: Site specific information, see V7 – Table H1 Procedure References

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

RU1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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 ODCM limits for 60 minutes or longer.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

(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) (2) Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer. (3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.	1) a. Valid ~~R~~reading on **ANY** ~~non-isolated~~ of the following effluent ~~path~~ radiation monitors greater than 2 times the ~~ODCM limits~~ value of the current high-alarm setpoint for 60 minutes or longer: RM-6509-1 (WTT Disch) RM-6521-1 (TB Sump) RM-6519-1 (SG Blowdown) RM-6473-1 (WT LIQ EFF) RM-6528-4 (WRGM rate) **AND** b. The discharge flow to the environment is not isolated within 60 minutes ~~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.~~ **OR** ~~3)~~ Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODCM limits for 60 minutes or longer. 2)
Difference /Justification: EAL RU1(2) deleted because it is redundant to EAL RU1(1)	
RU1.1: Site specific information, see V8 012_Table 03-15 - UFSAR WRGM Ranges and V32 UFSAR Table 11.5-1	

RU2: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
UNPLANNED loss of water level above irradiated fuel.	UNPLANNED loss of water level above irradiated fuel.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following: (site-specific level indications). **AND** b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors. (site-specific list of area radiation monitors)	(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: 1-SF-LI-2607 (Spent Fuel Pool Level) 1-SF-LI-2629 or 1-SF-LIT-2629-1 (Reactor Refuel Cavity Level) **AND** b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors: RM-6535-A-1, Containment Manipulator Crane RM-6535-B-1, Containment Manipulator Crane RM-6549-1, FSB Spent Fuel Range Low RM-6518-1, FSB Spent Fuel Range Hi
Difference /Justification	
CG1.1a: Site specific information, see V9 – SFP level **CG1.1b:** Site specific information, see V10 UFSAR Table 12.3-14 - CTMT Post-LOCA Range	

CG1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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 /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

(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).

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

AND

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

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

AND

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

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

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

(1) a. RVLIS Full Range < 55% (-141.5 in) for 30 minutes or longer.

AND

b. **ANY** indication from the Containment Challenge Table C2.

(2) a. Reactor vessel/RCS level cannot be monitored for 30 minutes or longer.

AND

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

RM-6535A-1 (Manipulator Crane) reading greater than 9500 mR/hr
RM-6535B-1 (Manipulator Crane) reading greater than 9500 mR/hr
Erratic source range monitor indication
UNPLANNED increase in Containment Sumps A or B levels of sufficient magnitude to indicate core uncovery.
Visual observation.

AND

c. **ANY** indication from the Containment Challenge Table C2.

Containment Challenge Table C2
• CONTAINMENT CLOSURE not established*
• Containment H_2 concentration $\geq 6\%$
• UNPLANNED increase in containment pressure

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

Difference /Justification

CG1.1a: Site specific information, see V11 EPCALC-06-04 - RVLIS Values
CG1.2b: Site specific information, see V10 UFSAR Table 12.3-14 – Manipulator Crane Monitor Range **and** V13 Containment Sumps
CG1.2c: Site specific information, see V14 H2 concentration in containment

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

CS1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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 /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level). (2) a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level). (3) a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [*PWR*] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications)	(1) a. CONTAINMENT CLOSURE not established. **AND** b. RVLIS Full Range < 63% (-101.9 in). (2) a. CONTAINMENT CLOSURE established. **AND** b. RVLIS Full Range < 55% (-141.5 in). (3) a. Reactor vessel/RCS level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following: RM-6535A-1 (Manipulator Crane) reading greater than 9500 mR/hr RM-6535B-1 (Manipulator Crane) reading greater than 9500 mR/hr Erratic source range monitor indication UNPLANNED increase in Containment Sumps A or B levels of sufficient magnitude to indicate core uncovery. Visual observation.
Difference /Justification	

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

CS1.1b & CS1.2b: Site specific information, see V11 EPCALC-06-04 - RVLIS Values **CS1.3b:** Site specific information, see V12 UFSAR Table 12.3-14 - CTMT Post-LOCA Range **and** V13 Containment Sumps

CA1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory.	Loss of reactor vessel/RCS inventory
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory as indicated by level less than (site-specific level). (2) a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored for 15 minutes or longer **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.	(1) Loss of reactor vessel/RCS inventory as indicated by RVLIS full range < 64% (-96.9 in). (2) a. Reactor vessel/RCS level cannot be monitored for 15 minutes or longer. **AND** b. UNPLANNED increase in Containment Sumps A or B levels due to a loss of reactor vessel/RCS inventory.
Difference /Justification	
CA1.1b: Site specific information, see V11 EPCALC-06-04 - RVLIS Values **CA1.2b**: Site specific information, see V13 Containment Sumps	

CA2: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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 emergency buses for 15 minutes or longer.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) Loss of **ALL** offsite and **ALL** onsite AC Power to (site-specific emergency buses) for 15 minutes or longer.	NOTE: For a bus to be considered energized from SEPS, both SEPS diesel generator sets must be functional. (1) Loss of **ALL** offsite and **ALL** onsite AC Power to **BOTH** AC emergency buses E5 AND E6 for 15 minutes or longer.
Difference /Justification	
Added NOTE for consideration of an additional non-safety power supply Supplemental Emergency Power System (SEPS)	

CA3: INITIATING CONDITIONS

NEI 99-01 Rev 6	Seabrook Station Nuclear Power Plant
Inability to maintain the plant in cold shutdown.	Inability to maintain the plant in cold shutdown.

Difference /Justification

None

THRESHOLDS

NEI 99-01 Rev 6

(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.

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

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

Seabrook Station Nuclear Power Plant

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

Table C1 - RCS Heat-up Duration Thresholds		
RCS Status	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
INTACT and reactor vessel ≥ -36 inches	Not applicable	60 minutes*
Not INTACT or reactor vessel < -36 inches	Established	20 minutes*
	Not Established	0 minutes
* If RHR is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

(2) UNPLANNED RCS pressure increase greater than 25 psig. (This EAL does not apply during water-solid plant conditions.)

Difference /Justification

CA3.1: Site specific information, see V15 Cold SD Temp Limit TS **and** V16 – Reduced Inventory
CA3.2: Site specific information, see V17 RCS Pressure range

CA6: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager **AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	(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 Other events with similar hazard characteristics as determined by the Shift Manager **AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.
Difference /Justification	

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

None

CU1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
UNPLANNED loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory for 15 minutes or longer.	UNPLANNED loss of reactor vessel/RCS inventory for 15 minutes or longer.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(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. (2) a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored. **AND** b. UNPLANNED increase in (site-specific sump and/or tank) levels.	(1) UNPLANNED loss of reactor coolant results in reactor vessel/RCS level less than a required lower limit of an operating band, specified by an operating procedure for 15 minutes or longer. (2) a. Reactor vessel/RCS level cannot be monitored. **AND** b. UNPLANNED increase in Containment Sump A or B level.
Difference /Justification	
CU1.2b: Site specific information, see V13 Containment Sumps	

CU2: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
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 emergency buses for 15 minutes or longer.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(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.	Note: For power restoration from the SEPS, both SEPS diesel generator sets must be functional. ~~(1~~1) a. AC power capability to BOTH AC emergency buses E5 AND E6 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. NOTE There are six power sources to consider: • 345 kV offsite power Line 369 • 345 kV offsite power Line 363 • 345 kV offsite power Line 394 • Emergency Diesel Generator A • Emergency Diesel Generator B • SEPS. For SEPS to be considered available, both SEPS diesel generator sets must be functional.
Difference /Justification	
Added NOTE to clarify that both SEPS constitute a single power source. Added NOTE containing table of AC power sources per EPFAQ 2015-15.	

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

CU3: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
UNPLANNED increase in RCS temperature.	UNPLANNED increase in RCS temperature.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit). (2) Loss of **ALL** RCS temperature and (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level indication for 15 minutes or longer.	(1) UNPLANNED increase in RCS temperature to greater than 200° F. (2) Loss of **ALL** RCS temperature and reactor vessel/RCS level indication for 15 minutes or longer.
Difference /Justification	
CU3.1: Site specific information, see V15 Cold SD Temp Limit TS	

CU4: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Loss of Vital DC power for 15 minutes or longer.	Loss of Vital DC power for 15 minutes or longer.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(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 105V on required Vital DC buses associated with the Protected Train for 15 minutes or longer. Train A - 11A and 11C Train B - 11B and 11D
Difference /Justification	
CU4.1: Site specific information, see V18 UFSAR 8.3.2 - DCV 105 limit	

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

CU5: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Loss of all onsite or offsite communications capabilities.	Loss of all onsite or offsite communications capabilities.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) Loss of **ALL** of the following onsite communication methods: (site-specific list of communications methods) (2) Loss of **ALL** of the following ORO communications methods: (site-specific list of communications methods) (3) Loss of **ALL** of the following NRC communications methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following onsite communication methods: In-Plant (PBX) Telephones Gai-Tronics Plant Radio System (2) Loss of **ALL** of the following ORO communications methods: Nuclear Alert System (NAS) Backup NAS ~~All~~ Control Room/TSC ~~plant~~ telephones ~~Cellular telephones~~ (3) Loss of **ALL** of the following NRC communications methods: Emergency Notification System (ENS) All plant telephones FTS telephones in the TSC ~~Cellular telephones~~
Difference /Justification	
Provided site specific communications methods	

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

Seabrook Station Nuclear Power Plant					
FG1 - Loss of any two barriers and Loss or Potential Loss of the third barrier.		**FS1 -** Loss or Potential Loss of any two barriers.		**FA1 -** Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.	
Difference /Justification					
None					
Fuel Clad Barrier		**RCS Barrier**		**Containment Barrier**	
Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
NEI 99-01 Rev 6					
1. RCS or SG Tube Leakage		**1. RCS or SG Tube Leakage**		**1. RCS or SG Tube Leakage**	
Not Applicable	A. RCS/reactor vessel level less than (site-specific level).	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: • UNISOLABLE RCS leakage **OR** • SG tube RUPTURE.	A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR**	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable

			B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).		
Seabrook Station Nuclear Power Plant					
Not Applicable	A. Core Cooling (C) CSF – ORANGE entry conditions met (NOTE 1)	A. An automatic or manual SI actuation is required by **EITHER** of the following: • UNISOLABLE RCS leakage **OR** • SG tube RUPTURE.	A. Operation of a second charging pump in the normal charging mode is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS Integrity (P) CSF – RED entry conditions met with RCS press > 300 psig. (NOTE 1)	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable
Difference /Justification					
Fuel Clad Barrier Potential Loss 1.A: Site specific information, see V20 CSFST Core Cooling **RCS Barrier Potential Loss 1.B:** Site specific information, see V21 CSFST Integrity					

FISSION PRODUCT BARRIER ICS/EALS

NEI 99-01 Rev 6					
2. Inadequate Heat Removal		**2. Inadequate Heat Removal**		**2. Inadequate Heat Removal**	
A. Core exit thermocouple readings greater than (site-specific temperature value).	A. Core exit thermocouple readings greater than (site-specific temperature value). **OR** B. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	Not Applicable	A. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	Not Applicable	A. 1. (Site-specific criteria for entry into core cooling restoration procedure) **AND** 2. Restoration procedure not effective within 15 minutes.
Seabrook Station Nuclear Power Plant					
A. Core Cooling (C) CSF – RED entry conditions met. (NOTE 1)	A. Core Cooling (C) CSF – ORANGE entry conditions met.(NOTE 1) **OR** B. Heat Sink (H) CSF – RED entry conditions met. (NOTE 1)	Not Applicable	A. Heat Sink (H) CSF – RED entry conditions met. (NOTE 1)	Not Applicable	A. Core Cooling (C) CSF – RED entry conditions met for 15 minutes or longer. (NOTE 1)
Difference /Justification					
Fuel Clad Barrier: Loss 2.A, Potential Loss 2.A and Containment Barrier Potential Loss 2.A: Site specific information, see V20 CSFST Core Cooling **RCS Barrier: Potential Loss 2.A:** Site specific information, see V22 CSFST Heat Sink					

FISSION PRODUCT BARRIER ICS/EALS

NEI 99-01 Rev 6					
3. RCS Activity / Containment Radiation		**3. RCS Activity / Containment Radiation**		**3. RCS Activity / Containment Radiation**	
A. Containment radiation monitor reading greater than (site-specific value). **OR** B. (Site-specific indications that reactor coolant activity is greater than 300 μCi/gm dose equivalent I-131).	Not Applicable	A. Containment radiation monitor reading greater than (site-specific value).	Not Applicable	Not Applicable	A. Containment radiation monitor reading greater than (site-specific value).
Seabrook Station Nuclear Power Plant					
A. Post LOCA Radiation Monitors RM 6576A-1 or RM 6576B-1 $\geq$ 95 R/hr. **OR** B. RCS activity > 300 uCi/gm Dose Equivalent I 131 as determined per Procedure CS0925.01, Reactor Coolant Post Accident Sampling.	Not Applicable	A. Post LOCA Radiation Monitors RM 6576A-1 or RM 6576B-1 $\geq$ 16 R/hr.	Not Applicable	Not Applicable	A. Post LOCA Radiation Monitors RM 6576A-1 or RM 6576B-1 $\geq$ 1,305 R/hr.

Difference /Justification					
All Barriers: Loss & Potential Loss 3.A: Site specific information, see V23 EPCALC-06-01 -Rad Values for Fission Product Barrier Matrix					
NEI 99-01 Rev 6					
4. Containment Integrity or Bypass		**4. Containment Integrity or Bypass**		**4. Containment Integrity or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Containment isolation is required **AND** **EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director udgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment pressure greater than (site-specific value) **OR** B. Explosive mixture exists inside containment **OR** C. 1. Containment pressure greater than (site-specific pressure setpoint) **AND** 2. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.
Seabrook Station Nuclear Power Plant					