NEXTera® ENERGY SEABROOK

October 27, 2016

10 CFR 50.90

SBK-L-16162

Docket No. 50-443

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

Seabrook Station

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"*

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)

In Reference 1, 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 2, the NRC staff determined that additional information is necessary to support the staff's continued technical review of the proposed EAL scheme change. The enclosures to this letter provide the requested additional information. The enclosed mark ups and clean pages supersede the corresponding pages in Reference 1.

The changes to the LAR provided in this letter do not alter the conclusion in Reference 1 that the change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with this change.

No new or revised commitments are included in this letter.

AX45
NRR

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 October 27, 2016.

Sincerely,

Eric McCartney

Eric McCartney
Site Vice President
NextEra Energy Seabrook, LLC

Enclosures:

Enclosure 1 Response to Request for Additional Information RegardingLicense 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"*

Enclosure 2 Markup of Affected Seabrook Station Emergency Action Levels – Initiating Conditions, Threshold Values and Basis

Enclosure 3 Clean Copy of Seabrook Station Emergency Action Levels – Initiating Conditions, Threshold Values and Basis

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

cc: NRC Region I Administrator
NRC Project Manager
NRC Senior Resident Inspector

Director Homeland Security and Emergency Management
New Hampshire Department of Safety
Division of Homeland Security and Emergency Management
Bureau of Emergency Management
33 Hazen Drive
Concord, NH 03305

Mr. John Giarrusso, Jr., Nuclear Preparedness Manager
The Commonwealth of Massachusetts
Emergency Management Agency
400 Worcester Road
Framingham, MA 01702-5399

Enclosure 1 to 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"*

Background

By letter dated February 27, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16068A128), NextEra Energy Seabrook, LLC (NextEra) submitted a license amendment request (LAR) to adopt the emergency action level schemes pursuant to Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," at Seabrook Station, Unit No. 1.

The NRC staff has determined that additional information provided below is necessary to complete the review.

RAI-Seabrook-1

Section 2.7, "Classification of Short-Lived Events," does not contain the guidance provided in Section 5.7 of NEI 99-01, Revision 6, which states, in part:

> If an event occurs that meets or exceeds an EAL, the associated ECL (emergency classification level) must be declared regardless of its continued presence at the time of declaration.

Please explain why this key guidance from NEI 99-01, Revision 6, was omitted, or revise accordingly.

NextEra Response

Section 2.7 of the Seabrook Station technical basis is revised to incorporate the referenced guidance statement from NEI 99-01, Revision 6, Section 5.7.

RAI-Seabrook-2

The technical basis discussion for RA3 [AA3] in NEI 99-01, Revision 6, states:

> This IC (initiation condition) 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.

The technical basis discussion for HA5 [HA5] in NEI 99-01, Revision 6, states:

> 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.

The proposed Table H1 includes "Equipment Vaults" as a plant room/area that require access to operate equipment as noted above. It is not clear to the NRC staff what required equipment is contained within the "Equipment Vaults," or if there are additional rooms/areas that are identified as "Equipment Vaults" that do not contain equipment, but require access to perform actions (e.g. operate equipment) necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown.

For EAL RA3 [AA3] and HA5 [HA5], please address the following:

a. Please clarify what required equipment is contained in the "Equipment Vaults" identified in Table H1. Additionally, please provide justification for using the potentially vague room/area designation of "Equipment Vaults" as this designation could potentially impact a timely and accurate classification, or revise accordingly.

b. Table H1 indicates that access to the containment is required in Operating Modes 3 and 4. Please explain why access is required to the containment building for Mode 3 and 4, operations or revise accordingly. This explanation should include: (1) a listing of the specific areas of the containment for which access is required in Operating Modes 3 and 4, and (2) what procedural requirements necessitate access for performing actions necessary to maintain normal plant operation or to perform normal plant cooldown and shutdown.

c. Table H1 indicates that access to the entire turbine building is required for Operating Modes 1, 2 and 3. Please explain why access is required to the entire turbine building for Operating Modes 1, 2 and 3 operations, or revise accordingly. This explanation should include: (1) a listing of the specific areas of the turbine building for which access is required in Operating Modes 1, 2 and 3, and (2) what procedural requirements necessitate access for performing actions necessary to maintain normal plant operation or to perform normal plant cooldown and shutdown.

NextEra Response

a. The term "Equipment Vaults" refers to the Residual Heat Removal (RHR)/Containment Building Spray (CBS) equipment vaults which are attached to the Primary Auxiliary Building. Operation of RHR is required for transition from modes 3 to 4 and modes 4 to 5 respectively per Operations Procedure OS1000.04, Plant Cooldown from Hot Standby to Cold Shutdown. Table H1 in EALs RA3 and HA5 is revised to replace the term "Equipment Vaults" with "RHR/CBS Equipment Vaults".

b. Access is required to Containment levels 0 to -26 in modes 3 and 4 to put RHR in service per Operations Procedures OS1013.03 and OS1013.04, RHR Train A and RHR Train B Startup and Operation. Prior to placing RHR in service for a scheduled plant shutdown, ultrasonic testing (UT) at RHR piping sample points to verify water solid conditions is required to be conducted per surveillance procedure OX1456.02, ECCS Monthly System Verification. Prerequisite 2.1.18 of OS1013.03 states, "If "A" RHR train is being placed in service as part of a scheduled plant shutdown, the "A" RHR lines have been verified water solid by performing ultrasonic testing on point RH-3 per PM ECCS-UT-PIPING. Consideration should be given to test "B" train, point RH-5, at the same time." UT requires access to the loop 4 entry at level -26' to access the ladder to reach test point RH 5 at the -10' level downstream of RC-V-87 (RHR train B suction isolation valve). UT also requires access to the loop 1 entry at level -26' to access the ladder to reach test point RH 3 at the -10' level downstream of RC-V-22 (RHR train A suction isolation valve).

c. Access is required to Turbine Building 21' and 50' levels in modes 1, 2 and 3 for alignment of feedwater with the Startup Feed Pump (SUFP) per Operations Procedures OS1000.03, Plant Shutdown from Minimum Load to Hot Standby, and OS1000.04, Plant Cooldown from Hot Standby to Cold Shutdown. Procedure OS1035.02, Startup Feed Pump Operations, requires access to the SUFP area on the 21' level of the Turbine Building North when the condensate (CO) cleanup system is not in service. When placing CO cleaning in service per procedure ON1034.09, Condensate Cleanup System Operation, and realigning SUFP suction to CO cleaning, access is required to the 21' level of the Turbine Building East along the condensers and to the 50' level of the Turbine Building northeast area near the CO cleanup system filters. The use of the CO cleanup system is preferred for conservation of water inventory if the main condenser steam dumps are available for use, i.e., the main steam isolation valves are not closed and condenser vacuum is maintained. Per OS1021.01, Steam Generator Blowdown System Operation, realignment of the flash tank vapor from FW-E-23C to the main condenser requires access to the 50' level of the Turbine Building. Table H1 in EALs HA5 and RA3 is revised to specify the 21 ft elevation and the 50 ft elevation of the Turbine Building as areas requiring access in Operating Modes 1, 2 and 3.

Further review of Table H1 resulted in identification of additional areas that are not required to access equipment as required by initiating conditions RA3 and HA5. These areas are non-essential switchgear room, steam and feedwater pipe chases, and -31 ft elevation of the Waste Process Building. Accordingly, Table H1 is revised to delete these areas.

RAI-Seabrook-3

For RU1 [AU1], EAL 1, the assessment criteria is based on one of the listed radiation monitors being greater than 2 times the offsite dose calculation manual (ODCM) limits. In addition to providing a list of site-specific monitors, the developer's guidance in NEI 99-01, Revision 6, states:

> Radiation monitor readings should reflect values that correspond to a radiological release exceeding 2 times a release control limit.

Please explain how an assessment of this EAL can be performed in a timely and accurate manner, without including instrument values that represent 2 times the ODCM limits, or revise accordingly.

NextEra Response

The ODCM values are conservatively based on an isotopic mix that will vary over time and may change depending on plant operations (e.g., a dry fuel storage campaign). The alarms for the radiation monitors listed in EAL RU1, with the exception of the WRGM alarm, are set at levels that are well below the ODCM value. The WRGM alarm is set at the ODCM value. If a radiation monitor alarm comes in, the operating staff will enter the applicable AOP (OS1252.01, Process or Effluent High Radiation, or OS1252.02, Airborne High Radiation) and will direct the on-shift Chemistry Technician to validate the alarm using Chemistry procedure CS0905.10, Chemistry Response to RDMS or Waste Gas Oxygen Monitor Failure or Alarm. CS0905.10 contains instructions for the on-shift Chemistry Technician for obtaining and analyzing applicable samples in the event any of the monitors identified in RU1 is in alert or high alarm. Values that correspond to 2X the ODCM limit are identified in CS0905.10. CS0905.10 directs the on-shift Chemistry Technician to notify the Shift Manager as soon as possible that the EAL may apply. The applicable AOP also directs the operating staff to evaluate the EALs if limits are exceeded as reported by the on-shift chemistry technician. Therefore a monitor alarm provides ample time for the on-shift operating staff and chemistry personnel to verify a monitor level and compare it to the ODCM value. Procedures and processes are in place to validate a radiation monitor alarm and determine if the alarm value meets the EAL threshold in a timely manner.

RAI-Seabrook-4

Please provide justification for not including power supply tables for EALs MA1 [SA1], MU1 [SU1] and CU2 [CU2], based on NRC staff resolution provided in Emergency Preparedness Frequently Asked Question (EPFAQ) No. 2015-15 (ADAMS Accession No. ML16166A191), or revise accordingly.

NextEra Response

Per EPFAQ No. 2015-15, a table of AC power sources that is included as a note in the current EAL SA5 will be added to NEI 99-01 Revision 6 EALs MA1and CU2. EPFAC No. 2015-15 says that a table of power sources is expected for MA1 [SA1] and CU2 only. MU1 concerns loss of offsite AC power sources only and therefore does not require the power supply table.

SA5 Note:

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.

RAI-Seabrook-5

For EALs MG8 [SG8], MG1 [SG1], MS1 [SS1] and CA2 [CA2], please provide justification for including a discussion related to a specific power source that could compel a decision-maker to make a declaration, even though mitigation strategies are effective, or revise accordingly.

NextEra Response

EALs MG8, MG1, MS1, MA1 and CA2 include a discussion of the Supplemental Emergency Power System (SEPS) that can supply power to emergency buses E5 or E6 in the event of the loss of offsite power and the failure of both emergency diesel generators to start and load. Each of the referenced EALs contains a note that says "For a bus to be considered energized from SEPS, both SEPS diesel generator sets must be functional." The basis section for each of the EALs contains the statement "For power restoration from the SEPS, both SEPS diesel generator sets must be functional." SEPS Loading Calculation 9763-3-ED-00-02-F provides the basis for requiring both SEPS diesel generator engines to be functional in order to be credited for supplying power to an emergency bus. The calculation shows that the required load is greater than the capacity of one SEPS generator engine (2640 KW.) This calculation does not take into account the starting current required by various equipment.

RAI-Seabrook-6

For EALs CU5 [CU5] and MU6 [MU6], please address the following:

a. Criteria lists "all plant telephones" as an acceptable communication method. This could imply that an EAL would not have to be declared as long as there was at least one functioning telephone on site. Please provide justification that supports the use of "all plant telephones," which addresses how this condition could be assessed in a timely and accurate manner.

b. Criteria lists cellular telephones as an acceptable method of communication for offsite communications. As stated in NEI 99-01, Revision 6, communication methods with the offsite response organizations and the NRC should be "...described in the site Emergency Plan." Section 7 of the Seabrook Site Emergency Plan, which describes communication methods, does not include cellular phones. Please provide justification for listing cellular phones as a method of communication, or revise accordingly.

NextEra Response

a. The term "All plant telephones" in EALs 2 and 3 of CU5 and MU6 is replaced by "Control Room/TSC Telephones".

b. Cellular telephones is deleted from the list of communications methods in EALs 2 and 3 of CU5 and MU6.

RAI-Seabrook-7

For the fuel clad and reactor coolant system (RCS) fission product barriers, RED entry conditions for the heat sink critical safety function (CSF) are used as a threshold for a potential loss of the barrier. However, the NEI 99-01, Revision 6, guidance states:

> In accordance with EOPs (emergency operating plans), 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.

This guidance is included in the barrier threshold basis discussions; however, it is not included in the relevant barrier thresholds.

Please explain why the NEI 99-01, Revision 6, guidance concerning making classifications for heat sink conditions when operators intentionally reduce heat removal capability, in accordance with EOPs, is not included in the fission product barrier thresholds, or revise accordingly.

NextEra Response

Current Seabrook Station Emergency Response Procedure ER 1.1, Classification of Emergencies, contains a discussion of the proper use of critical safety function status trees (CSFSTs) for emergency classification (i.e., non-green CSFST must represent a true challenge to the CSF for emergency classification purposes). This discussion is retained in the revised ER 1.1 procedure that incorporates the NEI 99-01 Revision 6 EALs. A note is added to the Fission Product Barrier Table for EALs FG1, FS1 and FA1 that refers to the discussion of proper use of CSFSTs for emergency classifications.

RAI-Seabrook-8

Concerning EAL HG1 [HG1], NRC staff resolution to EPFAQ 2015-13 (ADAMS Accession No. ML16166A366) was recently approved, which provides guidance that could be used, if deemed appropriate, to meet the intent of HG1 [HG1]. Please consider EPFAQ 2015-13 and revise EAL HG1 [HG1] if deemed appropriate, to reflect latest staff clarification of NEI 99-01, Revision 6 guidelines.

NextEra Response

Per EPFAQ 2015-13, EAL HG1 is deleted. Because this is a deviation from NEI 99-01, Revision 6, the table of deviations and differences is revised to include the justification for deletion of EAL HG1.

RAI-Seabrook-9

EAL HU4 [HU4] (2) in NEI 99-01, Revision 6, states:

> Receipt of a single fire alarm (i.e., no other indications of a FIRE).

The NEI 99-01, Revision 6, technical basis for HU4 [HU4] (2) further states:

> 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.

The proposed HU4 [HU4] (2) includes an exception for the containment based on the following note:

> A containment fire alarm is considered valid upon receipt of an actuated alarm on CP-376, combined with any of the following:
> - CP 376 panel - Multiple Zones Actuated
> - Plant Equipment - Spuriously Operating
> - Containment Temperature - Increasing
> - Containment Particulate Radiation - Increasing

Please provide further justification for the apparent deviation from the NRC-endorsed guidance provided by NEI 99-01, Revision 6, for the receipt of a single fire alarm. For example, this could potentially cause confusion with a declaration under EALs MA9 [SA9] and CA6 [CA6], where a containment fire causes spurious operation of equipment, e.g., is a rise in containment temperature or spurious operation of equipment to be considered as indications of degraded performance per MA9 [SA9] and CA6 [CA6]?

NextEra Response

Next Era proposes to make an exception in EAL HU4 (2) to exclude containment in modes 1 and 2 because accessing containment within 30 minutes to verify the status of a single alarm is a challenge, particularly in modes 1 and 2 when containment integrity is set and personnel safety concerns would preclude entry into certain areas of containment. There are areas within containment where fire detectors are located that would be inaccessible during these modes due to elevated radiation levels. Based on prior experience, if containment were to be included in EAL HU4 (2) during modes 1 and 2, the potential would exist for an inordinate unneeded number of Notification of Unusual Event emergency classifications and subsequent retractions.

Seabrook Station's containment building contains 137 individual Pyrotronics detectors distributed over 9 zones. The first 4 zones provide detection for the 0' elevation; the other 5 zones provide detection for the -26' elevation. When a detector alarms, the zone alarm for the zone in which the detector is located will actuate on fire panel FP-CP-376. 137 individual fire detectors are an unusually large number that significantly increases the potential of a spurious alarm. The 137 fire detectors in the Seabrook Station containment building is approximately 4.5 times the average number of containment building fire detectors in other NextEra nuclear power plants.

Actuation of more than one zone on FP-CP-376 is the most reliable indication of a valid fire detector alarm because of the volume of air flow throughout the containment building. Due to construction of the intermediate floors and multiple openings in the floors it can be expected that smoke would migrate throughout containment. There are six Containment Air Handling (CAH) cooling units located on the 0' elevation of the containment building. Five of the CAH cooling units are normally operating at any given time to cool the containment. Each cooling unit discharges approximately 56,000 CFM into the common air distribution system. The units draw return air into each end of the unit. This constant flow of air (approximately 280,000 CFM) would draw any smoke towards the cooling units past the installed detectors thus affecting multiple zones. More than one zone actuated on FP-CP-376 is therefore the most reliable indication of a valid alarm and accurately meets the criteria of EAL HU4 (1). Verification of a single containment fire alarm that is likely to be spurious does not warrant the potential elevated exposure risks associated with an emergency entry of containment in modes 1 and 2. Therefore, Seabrook Station proposes to make EAL HU4(2) applicable to a single fire alarm in containment in Modes 3, 4, 5 and 6.

The note containing criteria for a valid containment fire alarm that would be applicable to HU4(1) during modes 1 and 2 is revised to read "A containment fire alarm is considered valid upon receipt of multiple zones (more than 1) actuated on CP-376 panel." The alternate indications of spurious equipment actuation, increasing containment temperature, and increasing particulate radiation in containment are removed from the note to preclude potential confusion with the degraded safety equipment EALs CA6 and MA9.

RAI-Seabrook-10

EAL HU4 [HU4] (4) in NEI 99-01, Revision 6, states:

> A FIRE within the plant or ISFSI (for plants with an ISFSI outside the plant Protected Area) PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

The proposed EAL HU4 [HU4] (4) does not include the independent spent fuel storage installation (ISFSI) (referred to as dry fuel storage facility).

Please explain why the dry fuel storage facility was not included for fires that require an offsite fire response to extinguish, or revise accordingly.

NextEra Response

EAL HU4(4) is revised to identify the Dry Fuel Storage Facility.

RAI-Seabrook-11

For EALs MU5 [SU5], MA5 [SA5], and MS5 [SS5], a power level (<5%) was added to the EALs. The intent of NEI 99-01, Revision 6, is align the above EAL classifications with site-specific EOP criteria of a successful reactor shutdown, as the consistency between EALs and EOPs would benefit the decision-makers by providing consistent criteria. The power level provided in the NEI 99-01, Revision 6, developer notes is an example that represents a typical EOP indication for a generic power plant.

Please consider either using either the same EOP reactor shutdown criteria that the operators use in either the EOPs or operator training, or consider using wording similar to the guidance in NEI 99-01, Revision 6.

NextEra Response

The reference to "neutron flux <5%" is removed from EALs MU5, MA5 and MS5. The wording in NEI 99-01, Revision 6, for SU5, SA5 and SS5 is used for these EALs instead.

RAI-Seabrook-12

For EAL MS5 [SS5], the second paragraph in the technical basis includes a discussion that classifications from MS5 [SS5] may be at a higher level than what would be determined by the fission product barrier recognition category. Although this may be true for some licensees, the Seabrook fission product barrier recognition category for either core cooling or heat sink CSF red entry conditions met would result in a site area emergency based solely on the fission product barrier recognition category. Please provide an explanation for including a discussion that does not appear to be specific to Seabrook, or revise accordingly.

NextEra Response

The second paragraph of the technical basis for EAL MS5 is deleted.

RAI-Seabrook-13

EAL MA1 [SA1] (1) in NEI 99-01, Revision 6, states:

a. AC (alternating current) 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 a loss of all AC power to SAFETY SYSTEMS.

For EAL MA1 [SA1], the condition that any additional single power source will result in a loss of all AC power to SAFETY SYSTEMS was removed from the proposed EALs as being redundant to the condition that AC power capability to both AC emergency buses E5 and E6 is reduced to a single power source for 15 minutes or longer. Although the conditions provided by NEI 99-01, Revision 6, both include the term power source, they are not redundant.

Please explain, in greater detail, why the condition, "Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS," was removed from the proposed EAL MA1 [SA1], or revise accordingly.

NextEra Response

EAL MA1 is revised to add the condition, "Any additional single power source failure will result in a loss of AC power to SAFETY SYSTEMS."

RAI-Seabrook-14

For EAL MA2 [SA2], please address the following:

a. As proposed, all core exit temperatures and all but one RCS temperatures would not require a classification. Depending on the nature of the transient, an RCS temperature indication may or may not provide an accurate assessment of core conditions.

Please justify, including RCS temperature as an alternative to core exit temperatures or revise accordingly.

b. The Seabrook core cooling critical safety function status tree (CSFST) specifically uses reactor vessel level indication system (RVLIS) to assess the Core Cooling CSFST. However, the proposed EAL MA2 [SA2] uses pressurizer level. Depending on the nature of the transient, pressurizer level indication may or may not provide an accurate assessment of core conditions.

Please provide justification for not using RVLIS to determine RCS level for EAL MA2 [SA2], or revise accordingly.

NextEra Response

a. EAL MA2 (1) a. is revised to delete RCS Temperature and utilize Core Exit Temperature as the indicated parameter.

b. EAL MA2 (1) a. is revised to delete Pressurizer Level and utilize RCS Level as the indicated parameter.

Enclosure 2 to SBK-L-16162

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

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 ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include a failure of the reactor protection system to automatically scram/trip the reactor followed by a successful manual scram/trip or an earthquake.

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
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

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.

RA3

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
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

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.

~~(1)~~

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

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

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
~~All~~Control Room/TSC ~~plant~~ telephones
~~Cellular telephones~~

OR

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

Emergency Notification System (ENS)
~~All~~Control Room/TSC ~~plant~~ telephones
FTS telephones in the TSC
~~Cellular telephones~~

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.

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
~~All~~Control Room/TSC ~~plant~~ telephones
~~Cellular telephones~~

OR

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

Emergency Notification System (ENS)
~~All~~Control Room/TSC ~~plant~~ telephones
FTS telephones in the TSC
~~Cellular telephones~~

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.

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. (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)

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 $\geq$ 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. .
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 $\geq$ 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

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 ~~an actuated alarm~~ multiple zones (more than 1) actuated on CP-376 panel.~~, combined with any of the following:~~
    - ~~CP 376 panel — Multiple Zones Actuated~~
    - ~~Plant Equipment — Spuriously Operating~~
    - ~~Containment Temperature — Increasing~~
    - ~~Containment Particulate Radiation — Increasing~~

(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 Vault	

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 1and 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.

ECL: Site Area Emergency

Initiating Condition: Inability to shutdown the reactor ~~to neutron flux < 5%~~ 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 ~~to neutron flux < 5%~~.

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: Alert

Initiating Condition: Automatic or manual trip fails to shutdown the reactor ~~to neutron flux < 5%~~, 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 ~~to neutron flux < 5%~~.

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 ~~r~~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: Notification of Unusual Event

Initiating Condition: Automatic or manual trip fails to shutdown the reactor~~to neutron flux < 5%.~~

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 <u>automatic</u> trip did not shutdown the reactor~~to neutron flux < 5%~~.

AND

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

OR

(2) a. A <u>manual</u> trip did not shutdown the reactor~~to neutron flux < 5%~~.

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.

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
~~Pressurizer~~ RCS Level
RCS Pressure
Core Exit ~~or RCS~~ 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.

Enclosure 3 to SBK-L-16162

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

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 ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include a failure of the reactor protection system to automatically scram/trip the reactor followed by a successful manual scram/trip or an earthquake.

HA5

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
Switchgear Rooms Essential	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.

RA3

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
Switchgear Rooms Essential	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

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

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

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.

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.

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. (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)

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

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 Vault	

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 1and 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.

MS5

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.

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: 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: 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 <u>automatic</u> 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 <u>manual</u> 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.

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.

Enclosure 4 to SBK-L-16162

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

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

<table>
<tr>
<td>

(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)

</td>
<td>

(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	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

</td>
</tr>
<tr>
<td colspan="2">

Difference /Justification

</td>
</tr>
<tr>
<td colspan="2">

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

</td>
</tr>
</table>

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) 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) 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.	

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 Control Room/TSC telephones (3) Loss of **ALL** of the following NRC communications methods: Emergency Notification System (ENS) All plant telephones FTS telephones in the TSC
Difference /Justification	
Provided site specific communications methods	

INDEPENDENT SPENT FUEL STORAGE FACILITY (ISFSI) ICS/EALS

EU1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Damage to a loaded cask CONFINEMENT BOUNDARY.	Damage to a loaded cask CONFINEMENT BOUNDARY.
Difference /Justification	
None	
THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.	(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
Difference /Justification	
Added NOTE pulled from the basis allowing calculating surface dose from a distance dose **EU1.1:** Site specific information, see V19 NUHOMS HSM Dose Rates Technical Specification	

			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					

Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Containment isolation is required **AND** **EITHER** of the following: 3. Containment integrity has been lost based on STED/SED judgment. **OR** 4. 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. Cnmt. hydrogen concentration ≥ 6% **OR** C. 1. Containment pressure > 18 psig **AND** 2. Less than one full train of Cnmt. Building Spray (CBS) is operating per design for 15 minutes or longer.
Difference /Justification					
Containment Barrier: Potential Loss 4.A: Site specific information, see V24 CSFST Containment **Containment Barrier: Potential Loss 4.B:** Site specific information, see V14 H2 concentration in containment **Containment Barrier: Potential Loss 4.C1:** Site specific information, see V25 Containment Spray Setpoint					
NEI 99-01 Rev 6					
5. Other Indications		**5. Other Indications**		**5. Other Indications**	
A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)	A. (site-specific as applicable)
Seabrook Station Nuclear Power Plant					

Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable
Difference /Justification					
None					
NEI 99-01 Rev 6					
6. Emergency Director Judgment		**6. Emergency Director Judgment**		**6. Emergency Director Judgment**	
A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.
Seabrook Station Nuclear Power Plant					
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					
Difference /Justification					
None					

HG1: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
HOSTILE ACTION resulting in loss of physical control of the facility.	DELETED
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision). **AND** b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity control • Core cooling [*PWR*] / RPV water level [*BWR*] • RCS heat removal **OR** 2. Damage to spent fuel has occurred or is IMMINENT.	DELETED
Difference /Justification	
HG1 deleted per staff resolution to EPFAQ 2015-13. HG1 conditions are bounded by initiating conditions RA2, RS2, RG2, RS1, RG1, HS1, HS6, HS7 and HG2.	

<table>
<tr>
<td>(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas:
(site-specific list of plant rooms or areas with entry-related mode applicability identified)
AND
b. Entry into the room or area is prohibited or impeded.</td>
<td>(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>
<tr><th colspan="2">Table H1</th></tr>
<tr><th>Area</th><th>Mode</th></tr>
<tr><td>Primary Aux Building
25 ft elevation
7 ft elevation
- 26 ft elevation</td><td>1, 2, 3, 4</td></tr>
<tr><td>Turbine Building
21 ft elevation
50 ft elevation</td><td>1, 2, 3</td></tr>
<tr><td>Essential Switchgear Rooms</td><td>1, 2, 3, 4</td></tr>
<tr><td>Waste Process Building
25 ft elevation
-3 ft elevation</td><td>1, 2, 3</td></tr>
<tr><td>Containment</td><td>3, 4</td></tr>
<tr><td>RHR/CBS Equipment Vaults</td><td>3, 4</td></tr>
</table>
</td>
</tr>
<tr><td colspan="2">**Difference /Justification**</td></tr>
<tr><td colspan="2">**HA5.1b:** Site specific information, see V7 – Table H1 Procedure References</td></tr>
</table>

(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** of the following plant rooms or areas:
(site-specific list of plant rooms or areas)

(2) a. Receipt of a single fire alarm (i.e., no other indications of a FIRE).

AND

b. The FIRE is located within **ANY** of the following plant rooms or areas:
(site-specific list of plant rooms or areas)

AND

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

(3) A FIRE within the plant *or ISFSI [for plants with an ISFSI outside the plant Protected Area]* PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

(4) A FIRE within the plant *or ISFSI [for plants with an ISFSI outside the plant Protected Area]* PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

NOTE: A containment fire alarm is considered valid upon receipt of multiple zones (more than 1) actuated alarm on CP 376.

(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
Equipment Vault	

(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.

(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.

(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.

Difference /Justification

Added NOTE to clarify the containment fire alarm
HU4.1b: Site specific information, see V28 Verification of Fire Areas
HU4.2b: Containment is excepted in Modes 1 and 2 but is covered by the second note. EAL1 would be applicable. Entry into the Containment to perform verifications within 30 minutes in Modes 1 and 2 is a challenge.

MS5: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Inability to shut down the reactor causing a challenge to (core cooling [*PWR*] / RPV water level [*BWR*]) or RCS heat removal.	Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal.
Difference /Justification	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. All manual actions to shut down the reactor have been unsuccessful. **AND** c. **EITHER** of the following conditions exist: • (Site-specific indication of an inability to adequately remove heat from the core) • (Site-specific indication of an inability to adequately remove heat from the RCS)	(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) - RED entry conditions met.
Difference /Justification	
MS5.1c: Site specific information, see V20 CSFST Core Cooling **and** V22 CSFST Heat Sink	

MA1: 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 a loss of all AC power to SAFETY SYSTEMS.	(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	

SYSTEM MALFUNCTIONS

(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.

[BWR parameter list]	*[PWR parameter list]*
Reactor Power	Reactor Power
RPV Water Level	RCS Level
RPV Pressure	RCS Pressure
Primary Containment Pressure	In-Core/Core Exit Temperature
Suppression Pool Level	Levels in at least (site-specific number) steam generators
Suppression Pool Temperature	Steam Generator Auxiliary or 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 scram *[BWR]* / trip *[PWR]*
- ECCS (SI) actuation
- Thermal power oscillations greater than 10% *[BWR]*

(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
Levels 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
ECCS (SI) actuation

Difference /Justification

None

MA5: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Automatic or manual (trip [PWR] / scram [BWR]) fails to shut down the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	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.
Difference /Justification	
None	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	(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.
Difference /Justification	

<table>
<tr><th colspan="2">MU2: INITIATING CONDITIONS</th></tr>
<tr><td>NEI 99-01 Rev 6</td><td>Seabrook Station Nuclear Power Plant</td></tr>
<tr><td>UNPLANNED loss of Control Room indications for 15 minutes or longer.</td><td>UNPLANNED loss of Control Room indications for 15 minutes or longer.</td></tr>
<tr><td colspan="2">Difference /Justification</td></tr>
<tr><td colspan="2">None</td></tr>
<tr><th colspan="2">THRESHOLDS</th></tr>
<tr><td>NEI 99-01 Rev 6</td><td>Seabrook Station Nuclear Power Plant</td></tr>
<tr><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.
<table>
<tr><th>[BWR parameter list]</th><th>[PWR parameter list]</th></tr>
<tr><td>Reactor Power</td><td>Reactor Power</td></tr>
<tr><td>RPV Water Level</td><td>RCS Level</td></tr>
<tr><td>RPV Pressure</td><td>RCS Pressure</td></tr>
<tr><td>Primary Containment Pressure</td><td>In-Core/Core Exit Temperature</td></tr>
<tr><td>Suppression Pool Level</td><td>Levels in at least (site-specific number) steam generators</td></tr>
<tr><td>Suppression Pool Temperature</td><td>Steam Generator Auxiliary or Emergency Feed Water Flow</td></tr>
</table></td><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.
<table>
<tr><td>Reactor Power</td></tr>
<tr><td>RCS Level</td></tr>
<tr><td>RCS Pressure</td></tr>
<tr><td>Core Exit Temperature</td></tr>
<tr><td>Level in at least two steam generators</td></tr>
<tr><td>Steam Generator Emergency Feed Water Flow</td></tr>
</table></td></tr>
<tr><td colspan="2">Difference /Justification</td></tr>
<tr><td colspan="2">None</td></tr>
</table>

MU5: INITIATING CONDITIONS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor.	Automatic or manual trip fails to shutdown the reactor..
Difference /Justification	

THRESHOLDS	
NEI 99-01 Rev 6	**Seabrook Station Nuclear Power Plant**
(1) a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. (2) a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. **EITHER** of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. **OR** 2. A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.	(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.
Difference /Justification	

MU6: 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 Control Room/TSC plant telephones (3) Loss of **ALL** of the following NRC communications methods: Emergency Notification System (ENS) All plant telephones FTS telephones in the TSC
Difference /Justification	
Provided site specific communications methods	