ENCLOSURE

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 286, ADOPTION OF EMERGENCY ACTION LEVEL SCHEME PURSUANT TO NEI 99-01 REVISION 6, "DEVELOPMENT OF EMERGENCY ACTION LEVELS FOR NON-PASSIVE REACTORS"

In Reference 1, NextEra Energy Point Beach, LLC (NextEra) submitted a request for an amendment to revise the facility operating licenses for the Point Beach Nuclear Plant (PBNP) Units 1 and 2. Specifically, the proposed change involves revising the Emergency Plan for PBNP to adopt the Nuclear Energy Institute's (NEI's) revised Emergency Action Level (EAL) scheme described in NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," which has been endorsed by the NRC.

In Reference 2, the NRC staff requested additional information to complete its review of the requested amendment. Enclosure 1 provides the NextEra response to the NRC staff's request for additional information.

PBNP RAI-1

For proposed EALs RU1, RA1, RS1, and RG1, explain the purpose of including the flowrates for the associated ventilation alignments, and describe what a decision maker would do if the flowrates could not be determined to be as listed.

NextEra Response RAI-1

The supporting calculation assumes flowrates for the different ventilation system alignments and these flowrates are used in the calculation of threshold values for the different alignments. The flowrates were carried over from the calculation results tables to the EALs. The decision makers use only the radiation monitor reading for the applicable system alignment to make the classification. Therefore, the flowrates have been removed from the proposed EALs.

The updated EALs are provided in the Attachments to this submittal.

PBNP RAI-2

Please address the following for proposed EALs RA1, RS1, and RG1:

a. *Explain why the proposed radiation monitor setpoints are significantly higher than the current radiation monitor setpoints.*

b. *It was not clear to the NRC staff if PBNP could accurately assess offsite dose based on steam line radiation monitors [1(2)RE-231 and 1(2)RE-232] for the range of steam generator pressures that may exist following a wide range of events. Additionally, the atmospheric steam dump or steam generator safety valve may not be fully open, which for either of these conditions, could result in an unnecessary declaration of a General Emergency classification. Explain how the steam line radiation monitors 1(2)RE-231 and 1(2)RE-232 can provide an accurate indication of dose based on setpoint values alone or revise accordingly.*

NextEra Response RAI-2a

The previous EAL threshold values for RA1, RS1, and RG1 are not directly comparable to those currently in use and those proposed in this submittal. A standardized methodology for completion of these calculations was not available until NRC endorsement of NEI 99-01, Revision 5, which contained a new Appendix A, Basis for Radiological Effluent EALs. The earlier PBNP EAL threshold values had been determined using site dose projection software in use at that time and were based on assumed meteorology and release parameters.

Example input parameters previously used in calculation of threshold values were:

- Wind speed of 10 mph with stability class D
- Release duration of 4 hours used
- Source term was estimated from site accident dose projection software based on:
    - Release starting at 4 hours after reactor shutdown
    - No containment sprays
    - For SGTRs, water level below the U-tubes (reduced iodine retention fraction)

In contrast, the proposed EAL threshold values have been calculated using site-specific parameters consistent with the updated ODCM, and the latest endorsed guidance of NEI 99-01; notably:

- x/Q values are taken from the PBNP ODCM directly rather than calculated from meteorological inputs
- Source term nuclide specific activity fractions are based on annual activity release data consistent with the PBNP ODCM
- Dose conversion factors (DCFs) for early phase of nuclear accidents are taken from EPA 400-R-92-001
- Release duration of 1 hour used

These updated calculated values have been adopted in the current PBNP EALs, which were issued on March 8, 2017 as Revision 30 to the PBNP Emergency Plan, Appendix B, Emergency Classification. This revision was submitted to the NRC by NextEra under letter number NRC 2017-0015, dated March 17, 2017, titled "Report of Changes to Emergency Plan." In this submittal, NextEra stated that:

> *EP Appendix B was revised due to interdependent updates to the Offsite Dose Calculation Manual and Calculation 2013-0018 (EC 288117) resulting in a revision to the emergency action levels and also includes editorial changes.*
>
> *These changes were evaluated in accordance with 10 CFR 50.54(q)(3) and it was determined that these changes did not reduce the effectiveness of the PBNP Emergency Plan.*

NextEra Response RAI-2b

The 1(2)RE-231 and 1(2)RE-232 radiation monitors are designed to provide post-accident effluent monitoring in compliance with Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." These instruments monitor the main steam lines to determine if there are any significant radiological releases from this potential effluent pathway. The calculation of a single threshold value for Initiating Conditions (IC) RA1, RS1, or RG1 does require the use of multiple assumptions (e.g., primary to secondary leak rate, effluent pathway (e.g., safety valve(s) or atmospheric relief), and release rate) all of which could result in an incorrect assessment of these ICs in several credible scenarios. Additionally, these instruments (1(2)RE-231 and 1(2)RE-232) do not have the capability to quantify an effluent release rate and do not have an established ODCM methodology for determining effluent monitor alarm setpoints.

Because of these factors, and in consideration of the variable system flow rates that may exist during a steam generator tube rupture (SGTR) event, the use of a single calculated value for these instruments to determine if an EAL threshold is exceeded cannot be reliably used as the sole criteria for the declaration of ICs RA1, RS1, or RG1. Therefore, PBNP proposes to implement an EAL scheme that will not use the 1(2)RE-231 and 1(2)RE-232 radiation monitor threshold values as declaration criteria for ICs RA1, RS1, or RG1. The revise proposed EALs are shown in the Attachments to this submittal.

While this proposed change is a reduction of the current means available to assess whether the initiating conditions of RA1, RS1, or RG1 exist, multiple and diverse means of assessment do remain available to EAL decision-makers in the form of alternate EALs within these ICs. In the event of a radiological release from the steam line pathway, EAL decision-makers can determine that conditions exist to warrant a declaration of ICs RA1, RS1, or RG1 by means of real-time dose assessment using actual steam flow and meteorology at the time of the event, or by means of field survey results that exceed pre-determined threshold values (alternate EALs within the same IC). PBNP maintains a full dose assessment capability (Unified RASCAL Interface (URI)) that uses the 1(2)RE-231 and 1(2)RE-232 radiation monitors, along with actual steam line flow rates, to perform off-site dose projections. At PBNP, dose assessment is an on-shift capability performed by a minimum staffing assignee trained to fulfill the dose assessment function.

The proposed change to not use the 1(2)RE-231 and 1(2)RE-232 radiation monitor instrument values as declaration criteria for ICs RA1, RS1, or RG1 does not detract from the guidance in NEI 99-01, the requirements of 10 CFR 50.47(b)(4), or the standards in Appendix E to 10 CFR 50. Additionally, the proposed change does not represent an EAL scheme change.

PBNP RAI-3

PBNP proposed EAL RA3.1 includes the Central Alarm Station (CAS) and the Secondary Alarm Station (SAS) as threshold criteria. Typically only one of these areas is required. If the CAS and SAS can both provide access to areas required to assure safe plant operations, explain why EAL does not provide an "AND" logic to the CAS and SAS, or select the primary station as the threshold value as provided in accordance with endorsed guidance.

NextEra Response RAI-3

PBNP added the "AND" logic to the CAS and SAS listing in EAL RA3.1 to clarify that each location is capable of providing access to areas required to assure safe plant operations. Therefore, loss of both CAS AND SAS would be required to meet the threshold intent of an actual or potential substantial degradation of the level of safety of the plant.

The revised proposed EAL is shown in the Attachments to this submittal.

PBNP RAI-4

Tables in proposed EALs RA3 and HA5 contain areas that do not appear to require entry to either maintain normal operation or to shut down and cooldown the plant (e.g., plants typically can transition from Mode 1 to Mode 3 without being required to enter the turbine building.) Additionally, plants can typically open the reactor trip breakers from the control room.

Please verify all the listed rooms or areas are restricted to only those areas that contain equipment needed for safe operation or safe shutdown / cool-down, or revise accordingly consistent with endorsed guidance.

NextEra Response RAI-4

Following receipt of the RAI, a step-by-step analysis was performed that reviewed all PBNP procedures that contained critical operational activities conducted outside of the Control Room used to implement the normal at-power procedure and the procedures used to shut down and cooldown the plant. This analysis considered the "*rooms or areas that contain equipment which require a manual/local action*" as noted in the NEI 99-01, Revision 6 developer notes. The analysis included the following PBNP procedures:

- OP 3A: Power operation to Hot Standby
- OP 3B: Reactor Shutdown
- OP 3C: Hot Standby to Cold Shutdown
- OP 7A: Placing RHR System in operation
- OP 5D: Part 3: Preparation for Chemically Degassing the RCS
- OP 5D: Part 4: Degassing the RCS using the PZR and Letdown Gas Stripper

This analysis did not include rooms or areas in which actions of a contingent or emergency nature would be performed. (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). This analysis also did not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

The resulting list of rooms or areas that contain equipment needed for safe operation or safe shutdown / cool-down now reads:

SAFE OPS, S/D, C/D AREAS	
Area/Building	**MODE**
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. elev.	3 / 4
Pipeway 3, 8 ft. elev.	3 / 4
1/2B-32 MCC Area	4

This listing has been updated in both IC/EAL RA3 and HA5, as shown in the Attachments to this submittal, and a summary of the step-by-step analysis is provided in the supporting technical documentation of Attachment 4.

PBNP RAI-5

Proposed EALs CU1, CA1, CS1, and CG1, affecting RCS inventory, include only Containment Sump "A" level rise as an indication of RCS leakage. Previously approved PBNP EALs also included Waste Holdup Tank level rise as an indication of RCS leakage.

Please explain the basis for deleting Waste Holdup Tank level rise as an indication of RCS leakage, or revise to include sump and or tank indications that would be indicative of a RCS leak.

NextEra Response RAI-5

Waste Holdup Tank level rise as an indication of RCS inventory loss has been returned to proposed EALs CU1, CA1, CS1, and CG1. The revised proposed EALs are shown in the Attachment to this submittal.

PBNP RAI-6

The basis for proposed EALs CU2, SU1, and SA1 include the following:

> *Unit 1(2) offsite power sources include:*
>
> - *345 KVAC system supplying power to the 13.8 KVAC system and the 1 (2)X04 transformer*
> - *cross-tying with the opposite unit power supply*
> - *Power to the 1 (2)X -02 Auxiliary transformer through the 19 KV AC system and the 1(2)X-01 main step-up transformer*
>
> *The capability to cross-tie AC power takes credit for the redundant power source for this IC [initiating condition]. The inability to implement the cross-tie within 15 minutes warrants declaring a UE [Notification of Unusual Event].*

However, the staff could not determine which of the above sources of power could not be aligned within 15 minutes.

a. *Provide an explanation as to the meaning of "cross-tying with the opposite unit power supply," include how cross-tying is supported by abnormal operating procedures (AOPs) or emergency operating procedures (EOPs).*

b. *Explain what is meant by "[t]he inability to implement the cross-tie within 15 minutes warrants declaring a UE." Address why it appears that declaring a UE for not being able to perform the cross tie within 15 minutes would be appropriate if you did not have one power supply already available to an emergency bus, as this condition would warrant the declaration of an Alert or Site Area Emergency classification under EAL CA1 or SS1, respectively.*

c. *For EAL SA1, correct the UE reference in the following SA1 Basis statement (for an Alert declaration): "The inability to implement the cross-tie within 15 minutes warrants declaring a UE."*

NextEra Response RAI-6

The listing of power sources has been updated in EALs SA1 and CU2 to more clearly reflect the readily available normal power sources used in plant procedures.

Since EAL SU1 only deals with offsite power capability, no listing of normal power sources is provided in the Basis for that EAL.

a. Cross-tie of the opposite unit's power supply is controlled by procedure ECA-0.0, LOSS OF ALL AC POWER, steps 23-25 for Train A and steps 41-43 for Train B. For loss of a single emergency bus, procedure AOP-19A, TRAIN "A" SAFEGUARDS BUS RESTORATION (Unit 1, Train A example), steps 16-17 control use of the opposite unit's power to energize the emergency bus. Both procedures are provided as additions to Attachment 4 – Updated Supporting Technical Information as validation documents V36 and V37.

b. This statement has been removed from the proposed Basis of the applicable EALs.

c. This statement has been removed from the proposed Basis of EAL SA1.

The revised proposed EALs are shown in the Attachments to this submittal.

PBNP RAI-7

The basis for proposed EALs CA2, SS1, and SG1 include the following:

- *Unit 1(2) offsite power sources include:*
  - *345 KVAC 1(2)X-03 through the 13.8 KVAC system to the 1(2)X04 transformer*
  - *345 KVAC through the 19 KVAC system to the aux transformer 1(2)X-02*
- *Unit 1 (2) onsite power sources consist of:*
  - *emergency diesel generators*
  - *gas turbine generator*
  - *unit main turbine generator*
  - *power supplied from the opposite unit*

Considering that threshold values EALs CA2, SS1, and SG1 are a loss of all offsite and onsite AC power to the emergency buses, the above list of power sources is not required. Additionally, the inclusion of this table in the Basis discussion for EALs CA2, SS1, and SG1 could imply that a decision maker would not potentially make a declaration for a loss of all AC power because a power supply that is not included in the Basis discussion is providing power to the emergency bus.

Please explain why the list of power sources is provided in the Basis discussions for EALs CA2, SS1, and SG1, or revise according to remove list.

NextEra Response RAI-7

The additional information in the Basis for these proposed EALs CA2, SS1, and SG1 has been removed to prevent potential confusion to the decision makers.

The following additional statement has been added to the EALs CA2, SS1, SG1, and SG2 (SG8 in NEI 99-01, Revision 6) Basis discussions to clarify to the decision makers that the use of other than normal power supplies can be credited in the evaluation of these EALs:

> *If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.*

The revised proposed EALs are shown in the Attachments to this submittal.

PBNP RAI-8

Description of communications systems appears to be inconsistent with that described in the PBNP Emergency Plan.

a. *Explain the difference between a commercial phone system, general telephone lines and a private branch exchange (PBX).*

b. *Verify that communications systems listed in EALs are consistent with those provided in the PBNP Emergency Plan and emergency plan implementing procedures.*

NextEra Response RAI-8a

The PBNP onsite private branch exchange (PBX) system is located on the Unit 1 Turbine Deck area and provides telephone extensions to the Control Room and onsite emergency facilities. The site PBX is normally powered from a plant lighting circuit and has a battery backup power supply in the event of a loss of power to the lighting circuit. The commercial telephone lines are external lines not powered or fully switched by the site PBX. These lines are also not supplied with backup power from the PBX battery backup power supply. Links between the Commercial lines and the PBX are provided to allow interoperability and limited fail-over protection (e.g., eight in-plant extensions automatically take over eight Mishicot, Wisconsin commercial lines upon loss of all in-plant PBX system power).

The term "general telephone lines" was previously used as a generic term for any desktop telephone which was not part of a dedicated network such as the Two-Digit Dial Select circuit (aka NARS Phone) or FTS2000. This term is no longer used at PBNP and has been removed from the proposed EALs. The revised proposed EALs are shown in the Attachments to this submittal.

NextEra Response RAI-8b

PBNP staff has reviewed the matrix of emergency response communications provided in the PBNP Emergency Plan, Figure 7-1, as well as applicable implementing procedures and have updated the listing provided in EALs CU5 and SU7 to be consistent. The revised proposed EALs are shown in the Attachments to this submittal.

PBNP RAI-9

The proposed EALs CS1 and CG1 do not appear to have been developed in accordance with endorsed guidance. The current PBNP EALs CS1 and CG1 refer to reactor vessel level indications of 0% on LI-447/LI-447A and 20 feet Reactor Vessel Level Indicating System (RVLIS) Narrow Range; however, these levels were not included in the proposed EAL scheme. Additionally, no level was provided that was 6 inches below the bottom ID of the reactor coolant system loop.

a. Provide further justification for the removal of an EAL that relies on the RVLIS, or revise accordingly. (Note: NEI 99-01, Revision 6, developer notes provide guidance for indication that is "approximately the top of active fuel.")

b. Provide Reactor Coolant System (RCS) level indication available near the bottom ID of the RCS loop, or explain why this is not addressed.

c. Explain why an indication that is normally available while in shutdown cooling was not used to provide a site-specific RCS level for CS1 or revise accordingly.

NextEra Response RAI-9a, b, c

In the PWR Developer notes provided by NEI 99-01, Revision 6 for these EALS, the following guidance is provided:

> *If the availability of on-scale level indication is such that this level value can be determined during some shutdown modes or conditions, but not others, then specify the mode-dependent and/or configuration states during which the level indication is applicable.* ***If the design and operation of water level instrumentation is such that this level value cannot be determined at any time during Cold Shutdown or Refueling modes, then do not include EAL #2(1) (classification will be accomplished in accordance with EAL #3(2)).*** [emphasis added]

The Reactor Vessel Level Indicating System (RVLIS) Narrow Range is only calibrated for use during hot conditions. In the Cold Shutdown or Refueling modes, this instrument is only intended to be used for trend information and is not used for definitive Reactor Vessel level indication. Additionally, level indicators LI-447/LI-447A have a limited range, with 0% being the bottom of the Hot leg nozzle and a reading of 10% defined as the lowest usable on-scale reading. For these reasons, neither instrument is an appropriate indicator for use in these EALs. Therefore, the guidance cited above was followed and these EALs of each IC were not used in the PBNP EAL scheme.

Note that the specific Developer guidance provided by NEI 99-01, Revision 6 as quoted above was not provided by the previously NRC-endorsed revisions of NEI 99-01; therefore, previous EALs based on that prior guidance included additional indications which were not well suited for this application.

PBNP RAI-10

For the proposed fuel clad and RCS fission product barriers, RED entry conditions Critical Safety Function Status Tree (CSFST) for the heat sink are used as a threshold for a potential loss of either of these barriers. However, endorsed guidance states:

> *In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.*

Please explain why the endorsed 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 as this could result in an inaccurate EAL declaration, or revise accordingly.

NextEra Response RAI-10

The guidance cited in the RAI was left out of the Basis document for the two barrier Potential Loss thresholds during reformatting. This guidance has been added to the PBNP Fission Product Barrier Basis for both Fuel Clad Barrier Potential Loss 1.B and RCS Barrier Potential Loss 1.A. The revised proposed EAL Basis is shown in the Attachments to this submittal.

PBNP RAI-11

The current PBNP Fission Product Barrier (FPB) EAL FC4 Potential Loss includes reactor vessel level indications by RVLIS. Please explain why these indications are not used in proposed FPB EAL FC2

NextEra Response RAI-11

RVLIS was previously only used in the PBNP Fission Product Barrier thresholds as a site-specific piece of the Core Cooling Critical Safety Function Status (CSFS) Orange Path logic. In NEI 99-01 Revision 6, "Basis Information for PWR EAL Fission Product Barrier Table 9-F-3," the Developer Notes for Threshold Parameters and Values state that a plant can use the CSFS either in lieu of plant parameters or in addition to listing the component pieces of the CSFS logic. This guidance was not previously provided in prior revisions of NEI 99-01. Using this new guidance, PBNP only used the CSFS as the method for the decision maker to quickly determine the EAL status since the operators continually monitor CSFS.

Excerpt from NEI 99-01, Revision 6 showing this guidance is provided below:

The CSFST thresholds may be addressed in one of 3 ways:

1) *Not incorporated; thresholds will use parameters and values as discussed in the Developer Notes.*
2) *Incorporated along with parameter and value thresholds (e.g., a fuel clad loss would have 2 thresholds such as "CETs > 1200°F" and "Core Cooling Red entry conditions met".*
3) *Used in lieu of parameters and values for all thresholds.*

Option 3 of the guidance cited above was used by PBNP in the development of the proposed EALs; therefore, the individual components of the Core Cooling CSFS logic tree are not listed separately.

PBNP RAI-12

Proposed EAL HU4.2 excludes the Containment from required verification of existence of a fire within 30 minutes of a single alarm in Modes 1 and 2. Please provide a justification that supports the HU4 note that excludes the Containment from consideration.

NextEra Response RAI-12

This exclusion was identified in the Enclosure to PBNP License Amendment Request 286 as Proposed Deviation #2 from the NRC-endorsed guidance provided in NEI 99-01, Revision 6.

The explanation and justification previously provided is restated and editorially updated below.

Proposed Deviation #2

NextEra proposes to make an exception in EAL HU4.2 to exclude from classification a single fire alarm in Containment, during Modes 1 and 2 only. Accessing Containment within 30 minutes to verify the status of a single fire alarm presents a personal safety risk, particularly in these Modes when Containment integrity is set and personnel safety concerns would preclude entry into certain areas of the Containment structure. There are also areas within Containment where fire detectors are located that would be inaccessible during these Modes due to elevated radiation levels. Therefore, verification of a single Containment fire alarm that may or may not be spurious would involve elevated risks (both industrial and radiation safety) associated with an emergency entry of Containment in Modes 1 and 2. Therefore, NextEra proposes to make EAL HU4.2 applicable to a single fire alarm in Containment in all plant conditions other than Modes 1 or 2.

Based on prior industry and PBNP experience, if Containment were to be included in EAL HU4.2 during Modes 1 and 2, a high potential exists for an unwarranted number of Unusual Event emergency classifications based on single spurious fire alarms. Additionally, at PBNP the Containment buildings are designated as NFPA-805 Low Safety Significant (LSS) fire areas, each containing 40 Photoelectric smoke detectors across three levels. The detectors and Fire Alarm Control Panels (FACP) have "smart" technology incorporated into all FACPs. Panels FACP-007 and FACP-008 have the Containment smoke detectors attached to them, however all FACPs onsite show all alarming detectors, providing the operating crew information on the alarming indicator to quickly ascertain the location of the alarming indicator. If a detector actuates, the actuation drives an alarm in the Main Control Room. All FACPs display the information from the alarming detector as well as the Control Room "FireWorks" panel.

There are four Containment Fan Cooler (CFC) units located throughout the Containment building. Each CFC has one Containment Accident Fan and one Containment Cooling fan delivering a combined 58,000 CFM. Three of the four CFCs are operating at any given time to cool the Containment. The units draw return air into each end of the unit and discharge into a common header. This constant flow of air (approximately 174,000 CFM) would draw any smoke towards the cooling units past the installed detectors, thus initiating multiple smoke detector alarms. Actuation of more than one smoke detector on the FACP system is the most reliable indication of an actual fire because of high volumetric air flow throughout the Containment building increasing the probability of any actual fire being sensed by multiple detectors. Due to construction of the intermediate floors and multiple openings in the floors, it can be expected that smoke would migrate throughout Containment in a very short period and that multiple (2 or more) smoke detectors would alarm. Basing emergency classifications on receiving more than one (>1) smoke detector actuation on the FACP system is therefore the most reliable indication of a valid alarm and accurately meets the Initiating Condition of HU4, "FIRE potentially degrading the level of safety of the plant."

The structure of the proposed deviation for HU4 IC/EAL is modelled after Seabrook Station's adoption of NEI 99-01 Revision 6 containing a similar exception, which was approved by the NRC in a Safety Evaluation dated February 10, 2017 (ML16358A411). Consistent with the guidance in Regulatory Issue Summary (RIS) 2003-18, Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4, dated January 2003, it is reasonable to conclude that the changes proposed to EAL HU4.2 would be considered a Deviation from the formally endorsed guidance of NEI 99-01, Revision 6.

PBNP RAI-13

The basis for proposed EAL HS6 states that the Operations Manual assumes the earliest operator action will be taken from a remote shutdown location is 30 minutes. The endorsed guidance provides a typical time of 15 minutes or a time based on a site-specific fire response analysis. Please provide a site-specific analysis that supports a response time of 30 minutes, or revise accordingly in accordance with endorsed guidance.

NextEra Response RAI-13

PBNP choses to return to use of the default 15 minute time frame provided in the Developer Notes of the endorsed guidance. The revised proposed EAL is shown in the Attachments to this submittal.

PBNP RAI-14

EAL SU4 does not indicate whether or not the Technical Specification allowable limits, as described by the RU3 IC, include completing required actions within the completion times as provided by the Technical Specifications. Please clarify whether or not Technical Specification completion times should be considered when assessing RU3.

NextEra Response RAI-14

PBNP understands that this RAI contains a typographic error and was intended to discuss IC SU3 of NEI 99-01 Revision 6 (renumbered to SU4 at PBNP), and PBNP answers accordingly.

PBNP has revised the EAL statement as follows to ensure allowed completion times are correctly taken into account by the decision makers when assessing the EAL:

SU4.2 *Sample analysis indicates that a RCS Specific Activity value is greater than an allowable limit specified in Technical Specifications as indicated by* ***ANY*** *of the following conditions:*

a. Dose Equivalent I-131 greater than 50 μCi/gm

OR

b. Dose Equivalent I-131 greater than 0.5 μCi/gm but less than or equal to 50 μCi/gm for greater than 48 hours

OR

c. Dose Equivalent Xe-133 greater than 300 μCi/gm for greater than 48 hours

The revised proposed EAL is shown in the Attachments to this submittal.

PBNP RAI-15

The final two paragraphs in Basis for proposed EAL SG1 appear to provide direction "to give the Emergency Director a reasonable idea of how quickly the need to declare a General Emergency." In addition to inappropriately including potential procedural direction in the Basis discussion, this direction is not consistent with endorsed guidance.

Please remove the last two paragraphs from the SG1 basis discussion or provide justification that the wording, which is not consistent with endorsed guidance, could not influence and Emergency Director from prematurely declaring a General Emergency.

NextEra Response RAI-15

The last two paragraphs from the SG1 basis discussion have been removed. The revised proposed EAL is shown in the Attachments to this submittal.

ATTACHMENT 1

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 286, ADOPTION OF EMERGENCY ACTION LEVEL SCHEME PURSUANT TO NEI 99-01 REVISION 6,
"DEVELOPMENT OF EMERGENCY ACTION LEVELS FOR NON-PASSIVE REACTORS"

UPDATED REDLINE MARKUP OF NEI 99-01 REVISION 6

289 pages follow

~~NEI 99-01 [Revision 6]~~

~~Development of Emergency Action Levels for Non-Passive Reactors~~

~~November 2012~~

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~~NEI 99-01 [Revision 6]~~

~~Nuclear Energy Institute~~

Point Beach Emergency Action Levels Bases Document

TBD, 2018

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~~November 2012~~

~~*Nuclear Energy Institute, 1776 I Street N. W., Suite 400, Washington D.C. (202.739.8000)*~~

~~ACKNOWLEDGMENTS~~

~~This document was prepared by the Nuclear Energy Institute (NEI) Emergency Action Level (EAL) Task Force.~~

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~~NOTICE~~

~~EXECUTIVE SUMMARY~~

~~Federal regulations require that a nuclear power plant operator develop a scheme for the classification of emergency events and conditions. This scheme is a fundamental component of an emergency plan in that it provides the defined thresholds that will allow site personnel to rapidly implement a range of pre-planned emergency response measures. An emergency classification scheme also facilitates timely decision-making by an Offsite Response Organization (ORO) concerning the implementation of precautionary or protective actions for the public.~~

~~The purpose of Nuclear Energy Institute (NEI) 99-01 is to provide guidance to nuclear power plant operators for the development of a site-specific emergency classification scheme. The methodology described in this document is consistent with Federal regulations, and related US Nuclear Regulatory Commission (NRC) requirements and guidance. In particular, this methodology has been endorsed by the NRC as an acceptable approach to meeting the requirements of 10 CFR § 50.47(b)(4), related sections of 10 CFR § 50, Appendix E, and the associated planning standard evaluation elements of NUREG-0654/ FEMA-REP-1, Rev. 1,~~ ~~*Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*~~~~, November 1980.~~

~~NEI 99-01 contains a set of generic Initiating Conditions (ICs), Emergency Action Levels (EALs) and fission product barrier status thresholds. It also includes supporting technical basis information, developer notes and recommended classification instructions for users. Users should implement ICs, EALs and thresholds that are as close as possible to the generic material presented in this document with allowance for changes necessary to address site-specific considerations such as plant design, location, terminology, etc.~~

~~Properly implemented, the guidance in NEI 99-01 will yield a site-specific emergency classification scheme with clearly defined and readily observable EALs and thresholds. Other benefits include the development of a sound basis document, the adoption of industry-standard instructions for emergency classification (e.g., transient events, classification of multiple events, upgrading, downgrading, etc.), and incorporation of features to improve human performance. An emergency classification using this scheme will be appropriate to the risk posed to plant workers and the public, and should be the same as that made by another NEI 99-01 user plant in response to a similar event.~~

~~The individuals responsible for developing an emergency classification scheme are strongly encouraged to review all applicable NRC requirements and guidance prior to beginning their efforts. Questions concerning this document may be directed to the NEI Emergency Preparedness staff, NEI EAL task force members or submitted to the Emergency Preparedness Frequently Asked Questions process.~~

~~Finally, unique State and local requirements associated with an emergency classification scheme are not reflected in this guidance. Incorporation of these requirements may be performed on a case-by-case basis in conjunction with the appropriate ORO agency. Any such changes will require a review under the applicable sections of 10 CFR 50.~~

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~~DEVELOMENT OF~~ POINT BEACH EMERGENCY ACTION LEVELS ~~FOR NON-PASSIVE REACTORS~~BASIS DOCUMENT

1 REGULATORY BACKGROUND

1.1 OPERATING REACTORS

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

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

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

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

~~The above list is not all-inclusive and it is strongly recommended that scheme developers consult with licensing/regulatory compliance personnel to identify and understand all applicable requirements and guidance. Questions may also be directed to the NEI Emergency Preparedness staff.~~

~~1.2 PERMANENTLY DEFUELED STATION~~

~~NEI 99-01 provides guidance for an emergency classification scheme applicable to a permanently defueled station. This is a station that generated spent fuel under a 10 CFR § 50 license, has permanently ceased operations and will store the spent fuel onsite for an extended period of time. The emergency classification levels applicable to this type of station are consistent with the requirements of 10 CFR § 50 and the guidance in NUREG 0654/FEMA-REP-1.~~

~~In order to relax the emergency plan requirements applicable to an operating station, the owner of a permanently defueled station must demonstrate that no credible event can result in a significant radiological release beyond the site boundary. It is expected that this verification will confirm that the source term and motive force available in the permanently defueled condition are insufficient to warrant classifications of a Site Area Emergency or General Emergency. Therefore, the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) applicable to a permanently defueled station may result in either a Notification of Unusual Event (NOUE) or an Alert classification.~~

~~The generic ICs and EALs are presented in Appendix C, *Permanently Defueled Station ICs/EALs*.~~

~~1.3~~1.2 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

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

The generic ICs and EALs for an ISFSI are presented in Section 8, ISFSI ICs/EALs. IC E-~~H~~U1 covers the spectrum of credible natural and man-made events included within the scope of an ISFSI design. This IC is not applicable to installations or facilities that may process and/or repackage spent fuel (e.g., a Monitored Retrievable Storage Facility or an ISFSI at a spent fuel processing facility). In addition, appropriate aspects of IC HU1 and IC HA1 should also be included to address a HOSTILE ACTION directed against an ISFSI.

The analysis of potential onsite and offsite consequences of accidental releases associated with the operation of an ISFSI is contained in NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees*. NUREG-1140 concluded that the postulated worst-case accident involving an ISFSI has

insignificant consequences to public health and safety. This evaluation shows that the maximum offsite dose to a member of the public due to an accidental release of radioactive materials would not exceed 1 rem Effective Dose Equivalent.

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

~~1.4~~1.3 NRC ORDER EA-12-051

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

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

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

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

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

NEI 99-01, Revision 6, includes three EALs that reflect the availability of the enhanced spent fuel pool level instrumentation associated with NRC Order EA-12-051. These EALs are included within ~~existing~~ ICs ~~AA2~~RA2, ~~and new ICs AS2~~ RS2, and ~~AG2~~RG2. ~~Associated EAL notes, bases and developer notes are also provided.~~

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

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

~~1.5 APPLICABILITY TO ADVANCED AND SMALL MODULAR REACTOR DESIGNS~~

~~The guidance in this document primarily addresses commercial nuclear power reactors in the United States, operating or permanently defueled, as of 2012 (so called 1st and 2nd generation plant designs); however, it may be adapted to advanced non-passive designs (often referred to as 3rd generation plant designs) as well. Developers of an emergency classification scheme for an advanced non-passive reactor plant may need to propose deviations from the generic guidance to account for the differences in design parameters and criteria, and operating characteristics and capabilities, between 2nd and 3rd generation plants.~~

~~There are significant design and operating differences between large commercial nuclear power plants (of any generation) and Small Modular Reactors (SMRs) (e.g., differences in source term). For this reason, this document is not applicable to SMRs.~~

2 KEY TERMINOLOGY USED IN NEI 99-01

There are several key terms that appear throughout the ~~NEI 99-01~~EAL methodology. These terms are introduced in this section to support understanding of subsequent material. As an aid to the reader, the following table is provided as an overview to illustrate the relationship of the terms to each other.

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

2.1 EMERGENCY CLASSIFICATION LEVEL (ECL)

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

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

2.1.1 ~~Notification of Unusual Event~~Unusual Event (~~NOUE~~UE)[+]

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

~~[+] This term is sometimes shortened to Unusual Event (UE) or other similar site-specific terminology. The terms Notification of Unusual Event, NOUE and Unusual Event are used interchangeably throughout this document~~

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

2.1.2 Alert

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

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

2.1.3 Site Area Emergency

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

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

2.1.4 General Emergency (GE)

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

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

2.2 INITIATING CONDITION (IC)

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

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

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

~~Considerations for the assignment of a particular Initiating Condition to an emergency classification level are discussed in Section 3.~~

2.3 EMERGENCY ACTION LEVEL (EAL)

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

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

2.4 FISSION PRODUCT BARRIER THRESHOLD

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

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

- Fuel Clad
- Reactor Coolant System (RCS)
- Containment

Upon determination that one or more fission product barrier thresholds have been exceeded, the combination of barrier loss and/or potential loss thresholds is compared to the fission product barrier IC/EAL criteria to determine the appropriate ECL.

In some accident sequences, the ICs and EALs presented in the Abnormal Radiation Levels/ Radiological Effluent (~~A~~R) Recognition Category will be exceeded at the same

time, or shortly after, the loss of one or more fission product barriers. This redundancy is intentional as the former ICs address radioactivity releases that result in certain offsite doses from whatever cause, including events that might not be fully encompassed by fission product barriers (e.g., spent fuel pool accidents, design containment leakage following a LOCA, etc.).

3 DESIGN OF THE ~~NEI 99-01~~PBNP EMERGENCY CLASSIFICATION SCHEME

3.1 ASSIGNMENT OF EMERGENCY CLASSIFICATION LEVELS (ECLS)

An effective emergency classification scheme must incorporate a realistic and accurate assessment of risk, both to plant workers and the public. There are obvious health and safety risks in underestimating the potential or actual threat from an event or condition; however, there are also risks in overestimating the threat as well (e.g., harm that may occur during an evacuation). The ~~NEI 99-01~~PBNP emergency classification scheme attempts to strike an appropriate balance between reasonably anticipated event or condition consequences, potential accident trajectories, and risk avoidance or minimization.

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

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

- Assessments of the effects and consequences of different types of events and conditions
- ~~Typical~~ PBNP abnormal and emergency operating procedure setpoints and transition criteria
- ~~Typical~~ PBNP Technical Specification limits and controls
- ~~Radiological Effluent Technical Specifications (RETS)/~~Offsite Dose Calculation Manual (ODCM) radiological release limits
- Review of selected Updated Final Safety Analysis Report (UFSAR) accident analyses
- Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs)
- NUREG 0654, Appendix 1, *Emergency Action Level Guidelines for Nuclear Power Plants*
- Industry Operating Experience
- Input from ~~industry~~ PBNP subject matter experts ~~and NRC staff members~~

The following ECL attributes were created ~~by the Revision 6 Preparation Team~~ to aid in the development of ICs and Emergency Action Levels (EALs). ~~The team decided to include the attributes in this revision since they~~The attributes may be useful in briefing and training settings (e.g., helping an Emergency Director understand why a particular condition is classified as an Alert). ~~It should be stressed that developers not attempt to redefine these attributes or apply them in any fashion that would change the generic guidance contained in this document[1].~~

~~[1] The use of ECL attributes is at the discretion of a licensee and is not a requirement of the NRC. If a licensee chooses in incorporate the ECL attributes into their scheme basis document, it must be very clear that the NRC staff has not endorsed their acceptability or application for any purpose. In particular, the staff does not consider the~~

~~attribute statements to supersede the established ECL definitions. As a result, the use of the attributes as a basis for justifying EAL changes is unacceptable.~~

~~The attributes of each ECL are presented below.~~

3.1.1 ~~Notification of Unusual Event~~Unusual Event (~~NOUE~~UE)

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

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

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

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

3.1.2 Alert

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

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

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

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

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

3.1.3 Site Area Emergency (SAE)

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

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

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

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

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

3.1.4 General Emergency (GE)

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

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

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

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

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

3.1.5 Risk-Informed Insights

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

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

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

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

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

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

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

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

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

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

~~3.3 NSSS Design Differences~~

~~The NEI 99-01 emergency classification scheme accounts for the design differences between PWRs and BWRs by specifying EALs unique to each type of Nuclear Steam Supply System (NSSS). There are also significant design differences among PWR NSSSs; therefore, guidance is provided to aid in the development of EALs appropriate to different PWR NSSS types. Where necessary, development guidance also addresses unique considerations for advanced non-passive reactor designs such as the Advanced Boiling Water Reactor (ABWR), the Advanced Pressurized Water Reactor (APWR) and the Evolutionary Power Reactor (EPR).~~

~~Developers will need to consider the relevant aspects of their plant's design and operating characteristics when converting the generic guidance of this document into a site-specific classification scheme. The goal is to maintain as much fidelity as possible to the intent of generic ICs and EALs within the constraints imposed by the plant design and operating characteristics. To this end, developers of a scheme for an advanced non-passive reactor may need to add, modify or delete some information contained in this document; these changes will be reviewed for acceptability by the NRC as part of the scheme approval process.~~

~~The guidance in NEI 99-01 is not applicable to advanced passive light water reactor designs. An Emergency Classification Scheme for this type of plant should be developed in accordance with NEI 07-01, *Methodology for Development of Emergency Action Levels, Advanced Passive Light Water Reactors*.~~

~~3.4~~3.3 PBNP-SPECIFIC ORGANIZATION AND PRESENTATION OF GENERIC INFORMATION

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

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

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

The following information and guidance is provided for each IC:

- **ECL** – the assigned emergency classification level for the IC.
- **Initiating Condition** – provides a summary description of the emergency event or condition.
- **Operating Mode Applicability** – Lists the modes during which the IC and associated EAL(s) are applicable (i.e., are to be used to classify events or conditions).

- **~~Example~~ Emergency Action Level(s)** – Provides examples of reports and indications that are considered to meet the intent of the IC. ~~Developers should address each example EAL. If the generic approach to the development of an example EAL cannot be used (e.g., an assumed instrumentation range is not available at the plant), the developer should attempt to specify an alternate means for identifying entry into the IC.~~

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

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

- ~~**Developer Notes** - Information that supports the development of the site-specific ICs and EALs. This may include clarifications, references, examples, instructions for calculations, etc. Developer notes should not be included in the site's emergency classification scheme basis document. Developers may elect to include information resulting from a developer note action in a basis section.~~

- ~~**ECL Assignment Attributes** – Located within the Developer Notes section, specifies the attribute used for assigning the IC to a given ECL.~~

~~3.5~~3.4 IC AND EAL MODE APPLICABILITY

The ~~NEI 99-01~~ PBNP emergency classification scheme was developed recognizing that the applicability of ICs and EALs will vary with plant mode. For example, some symptom-based ICs and EALs can be assessed only during the power operations, startup, or hot standby/shutdown modes of operation when all fission product barriers are in place, and plant instrumentation and ~~safety systems~~SAFETY SYSTEMS are fully operational. In the cold shutdown and refueling modes, different symptom-based ICs and EALs will come into play to reflect the opening of systems for routine maintenance, the unavailability of some ~~safety system~~SAFETY SYSTEM components and the use of alternate instrumentation.

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

MODE APPLICABILITY MATRIX

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

~~Typical BWR Operating Modes~~

~~Power Operations (1): Mode Switch in Run~~

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

~~Hot Shutdown (3): Mode Switch in Shutdown, Average Reactor Coolant Temperature >200 °F~~

~~Cold Shutdown (4): Mode Switch in Shutdown, Average Reactor Coolant Temperature ≤ 200 °F~~

~~Refueling (5): Mode Switch in Shutdown or Refuel, and one or more vessel head bolts less than fully tensioned.~~

~~Typical PWR~~ PBNP OPERATING MODES

MODE	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥0.99	> 5	NA
2	Startup	≥0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥350
4	Hot Shutdown[(b)]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown [(b)]	< 0.99	NA	≤ 200
6	Refueling [(c)]	NA	NA	NA
N/A	Defueled	All fuel removed from the reactor vessel (full core offload during refueling or extended outage)		
	(a) Excluding decay heat (b) All reactor vessel head closure bolts fully tensioned. (c) One or more reactor vessel head closure bolts less than fully tensioned.			

~~Power Operations (1): Reactor Power > 5%, Keff ≥ 0.99~~

~~Startup (2): Reactor Power ≤ 5%, Keff ≥ 0.99~~

~~Hot Standby (3): RCS ≥ 350 °F, Keff < 0.99~~

~~Hot Shutdown (4): 200 °F < RCS < 350 °F, Keff < 0.99~~

~~Cold Shutdown (5): RCS < 200 °F, Keff < 0.99~~

~~Refueling (6): One or more vessel head closure bolts less than fully tensioned~~

~~Developers will need to incorporate the mode criteria from unit-specific Technical~~

~~Specifications into their emergency classification scheme. In addition, the scheme must also include the following mode designation specific to NEI 99-01:~~

~~Defueled (None):~~	~~All fuel removed from the reactor vessel (i.e., full core offload during refueling or extended outage).~~

4 ~~SITE-SPECIFIC~~PBNP SCHEME DEVELOPMENT ~~GUIDANCE~~

~~This section provides detailed guidance for developing a site-specific emergency classification scheme. Conceptually, the approach discussed here mirrors the approach used to prepare emergency operating procedures – generic material prepared by reactor vendor owners groups is converted by each nuclear power plant into site-specific emergency operating procedures. Likewise, the emergency classification scheme developer will use the generic guidance in NEI 99-01 to prepare a site-specific emergency classification scheme and the associated basis document.~~

~~It is important that the NEI 99-01 emergency classification scheme be implemented as an integrated package. Selected use of portions of this guidance is strongly discouraged as it will lead to an inconsistent or incomplete emergency classification scheme that will likely not receive the necessary regulatory approval.~~

4.1 GENERAL ~~IMPLEMENTATION GUIDANCE~~DEVELOPMENT PROCESS

~~The guidance in NEI 99-01 is not intended to be applied to plants "as-is"; however, developers should attempt to keep their site-specific schemes as close to the generic guidance as possible. The goal is to meet the intent of the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) within the context of site-specific characteristics – locale, plant design, operating features, terminology, etc. Meeting this goal will result in a shorter and less cumbersome NRC review and approval process, closer alignment with the schemes of other nuclear power plant sites and better positioning to adopt future industry-wide scheme enhancements.~~

~~When properly developed, the~~ The PBNP ICs and EALs ~~should~~ were developed to be unambiguous and readily assessable.

~~As discussed in Section 3, the generic guidance includes ICs and example EALs. It is the intent of this guidance that both be included in site-specific documents as each serves a specific purpose.~~ The IC is the fundamental event or condition requiring a declaration. The EAL(s) is the pre-determined threshold that defines when the IC is met. ~~If some feature of the plant location or design is not compatible with a generic IC or EAL, efforts should be made to identify an alternate IC or EAL.~~

~~If an IC or EAL includes an explicit reference to a mode dependent technical specification limit that is not applicable to the plant, then that IC and/or EAL need not be included in the site-specific scheme. In these cases, developers must provide adequate documentation to justify why the IC and/or EAL were not incorporated (i.e., sufficient detail to allow a third party to understand the decision not to incorporate the generic guidance).~~

Useful acronyms and abbreviations associated with the ~~NEI 99-01~~PBNP emergency classification scheme are presented in Appendix A, Acronyms and Abbreviations. ~~Site-specific entries may be added if necessary.~~

Many words or terms used in the ~~NEI 99-01~~PBNP emergency classification scheme have scheme-specific definitions. These words and terms are identified by being set in all capital letters (i.e., ALL CAPS). The definitions are presented in Appendix B, Definitions.

~~Below are examples of acceptable modifications to the generic guidance. These may be incorporated depending upon site developer and user preferences.~~

- ~~The ICs within a Recognition Category may be placed in reverse order for presentation purposes (e.g., start with a General Emergency at the left/top of a user aid, followed by Site Area Emergency, Alert and NOUE).~~
- ~~The Initiating Condition numbering may be changed.~~
- ~~The first letter of a Recognition Category designation may be changed, as follows, provided the change is carried through for all of the associated IC identifiers.~~
  - ~~R may be used in lieu of A~~
  - ~~M may be used in lieu of S~~

~~For example, the Abnormal Radiation Levels / Radiological Effluent category designator "A" (for Abnormal) may be changed to "R" (for Radiation). This means that the associated ICs would be changed to RU1, RU2, RA1, etc.~~

- ~~The ICs and EALs from Recognition Categories S and C may be incorporated into a common presentation method (e.g., one table) provided that all related notes and mode applicability requirements are maintained.~~
- ~~The ICs and EALs for Emergency Director judgment and security-related events may be placed under separate Recognition Categories.~~
- ~~The terms EAL and threshold may be used interchangeably.~~

~~The material in the Developer Notes section is included to assist developers with crafting correct IC and EAL statements. This material is not required to be in the final emergency classification scheme basis document.~~

4.2 CRITICAL CHARACTERISTICS

~~As discussed above, developers are encouraged to keep their site-specific schemes as close to the generic guidance as possible.~~ When crafting the scheme, ~~developers should satisfy themselves~~PBNP ensured that certain critical characteristics have been met. These critical characteristics are listed below.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are consistent with industry guidance; while the actual wording may be different, the classification intent is maintained. With respect to Recognition Category F, ~~a site-specific scheme must~~ PBNP includes a ~~some type of~~ user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. The user-aid logic ~~must be~~is consistent with the classification logic presented in Section 9.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are technically complete and accurate (i.e., they contain the information necessary to make a correct classification).
- EAL statements use objective criteria and observable values.
- ICs, EALs, Operating Mode Applicability and Note statements and formatting consider human factors and are user-friendly.
- The scheme facilitates upgrading and downgrading of the emergency classification where necessary.
- The scheme facilitates classification of multiple concurrent events or conditions.

4.3 INSTRUMENTATION USED FOR EALS

~~Instrumentation referenced in EAL statements should include that described in the emergency plan section which addresses 10 CFR 50.47(b)(8) and (9) and/or Chapter 7 of the FSAR. Instrumentation used for EALs need not be safety-related, addressed by a Technical Specification or ODCM/RETS control requirement, nor powered from an emergency power source; however, EAL developers should strive to~~ PBNP incorporated instrumentation that is reliable and routinely maintained in accordance with site programs and procedures. Alarms referenced in EAL statements ~~should be~~are those that are the most operationally significant for the described event or condition.

~~Scheme developers should ensure that specified values used as~~ EAL setpoints are within the calibrated range of the referenced instrumentation, and consider any automatic instrumentation functions that may impact accurate EAL assessment. In addition, EAL setpoint values ~~should~~ do not use terms such as "off-scale low" or "off-scale high" since that type of reading may not be readily differentiated from an instrument failure. ~~Findings and violations related to EAL instrumentation issues may be located on the NRC website.~~

~~4.4 PRESENTATION OF SCHEME INFORMATION TO USERS~~

~~The US Nuclear Regulatory Commission (NRC) expects licensees to establish and maintain the capability to assess, classify and declare an emergency condition promptly within 15 minutes after the availability of indications to plant operators that an emergency action level has been, or may be, exceeded. When writing an emergency classification procedure and creating related user aids, the developer must determine the presentation method(s) that best supports the end users by facilitating accurate and timely emergency classification. To this end, developers should consider the following points.~~

- ~~The first users of an emergency classification procedure are the operators in the Control Room. During the allowable classification time period, they may have responsibility to perform other critical tasks, and will likely have minimal assistance in making a classification assessment.~~
- ~~As an emergency situation evolves, members of the Control Room staff are likely to be the first personnel to notice a change in plant conditions. They can assess the changed conditions and, when warranted, recommend a different emergency classification level to the Technical Support Center (TSC) and/or Emergency Operations Facility (EOF).~~
- ~~Emergency Directors in the TSC and/or EOF will have more opportunity to focus on making an emergency classification, and will probably have advisors from Operations available to help them.~~

~~Emergency classification scheme information for end users should be presented in a manner with which licensed operators are most comfortable. Developers will need to work closely with representatives from the Operations and Operations Training Departments to develop readily usable and easily understood classification tools (e.g., a procedure and related user aids). If necessary, an alternate method for presenting~~

~~emergency classification scheme information may be developed for use by Emergency Directors and/or Offsite Response Organization personnel.~~

~~A wallboard is an acceptable presentation method provided that it contains all the information necessary to make a correct emergency classification. This information includes the ICs, Operating Mode Applicability criteria, EALs and Notes. Notes may be kept with each applicable EAL or moved to a common area and referenced; a reference to a Note is acceptable as long as the information is adequately captured on the wallboard and pointed to by each applicable EAL[+]. Basis information need not be included on a wallboard but it should be readily available to emergency classification decision-makers.~~

~~In some cases, it may be advantageous to develop two wallboards - one for use during power operations, startup and hot conditions, and another for cold shutdown and refueling conditions.~~

~~Alternative presentation methods for the Recognition Category F ICs and fission product barrier thresholds are acceptable and include flow charts, block diagrams, and checklist-type tables. Developers must ensure that the site-specific method addresses all possible threshold combinations and classification outcomes shown in the BWR or PWR EAL fission product barrier tables. The NRC staff considers the presentation method of the Recognition Category F information to be an important user aid and may request a change to a particular proposed method if, among other reasons, the change is necessary to promote consistency across the industry.~~

~~4.5 INTEGRATION OF ICS/EALS WITH PLANT PROCEDURES~~

~~A rigorous integration of IC and EAL references into plant operating procedures is not recommended. This approach would greatly increase the administrative controls and workload for maintaining procedures. On the other hand, performance challenges may occur if recognition of meeting an IC or EAL is based solely on the memory of a licensed operator or an Emergency Director, especially during periods of high stress.~~

~~Developers should consider placing appropriate visual cues (e.g., a step, note, caution, etc.) in plant procedures alerting the reader/user to consult the site emergency classification procedure. Visual cues could be placed in emergency operating procedures, abnormal operating procedures, alarm response procedures, and normal operating procedures that apply to cold shutdown and refueling modes. As an example, a step, note or caution could be placed at the beginning of an RCS leak abnormal operating procedure that reminds the reader that an emergency classification assessment should be performed.~~

~~[+] Where appropriate, the Notes shown in the generic guidance typically include the event/condition ECL and the duration time specified in the EAL. If developers prefer to have several ICs reference a common NOTE on a wallboard display, it is acceptable to remove the ECL and time criterion and use a generic statement. For example, a common NOTE could read "The Emergency Director should declare the emergency promptly upon determining that the applicable EAL time has been exceeded, or will likely be exceeded."~~

~~4.6 Basis Document~~

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

- ~~A site-specific Mode Applicability Matrix and description of operating modes, similar to that presented in section 3.5.~~
- ~~A discussion of the emergency classification and declaration process reflecting the material presented in Section 5. This material may be edited as needed to align with site-specific emergency plan and implementing procedure requirements.~~
- ~~Each Initiating Condition along with the associated EALs or fission product barrier thresholds, Operating Mode Applicability, Notes and Basis information.~~
- ~~A listing of acronyms and defined terms, similar to that presented in Appendices A and B, respectively. This material may be edited as needed to align with site-specific characteristics.~~
- ~~Any site-specific background or technical appendices that the developers believe would be useful in explaining or using elements of the emergency classification scheme.~~

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

~~Basis information should be readily available to be referenced, if necessary, by the Emergency Director. For example, a copy of the basis document could be maintained in the appropriate emergency response facilities.~~

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

~~4.7~~4.4 EAL/Threshold References to AOP and EOP Setpoints/Criteria

~~As reflected in the generic guidance,~~Some of the criteria/values used in several EALs and fission product barrier thresholds may be drawn from ~~a plant's~~PBNP's AOPs and EOPs. This approach is intended to maintain good alignment between operational diagnoses and emergency classification assessments. ~~Developers should verify that a~~Appropriate administrative controls are in place to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required.

~~4.8 DEVELOPER AND USER FEEDBACK~~

~~Questions or comments concerning the material in this document may be directed to the NEI Emergency Preparedness staff, NEI EAL task force members or submitted to the Emergency Preparedness Frequently Asked Questions process.~~

5 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

5.1 GENERAL CONSIDERATIONS

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

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

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

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

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

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

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

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

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

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

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.); the EAL and/or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. ~~The NEI 99-01~~This scheme provides the Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated into the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

5.2 Classification Methodology

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

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

5.3 Classification of Multiple Events and Conditions

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. For example:

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

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

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

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

5.4 Consideration of Mode Changes During Classification

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

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

barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

5.5 CLASSIFICATION OF IMMINENT CONDITIONS

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

5.6 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND DOWNGRADING

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

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

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

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

5.7 Classification of Short-Lived Events

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

5.8 Classification of Transient Conditions

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

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

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

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

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

This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

5.9 After-the-Fact Discovery of an Emergency Event or Condition

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

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

5.10 Retraction of an Emergency Declaration

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

6 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

~~Table AR-1: Recognition Category "AR" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~AU1~~RU1 ~~Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document)~~ODCM ~~limits for 60 minutes or longer.~~ ~~*Op. Modes: All*~~	~~AA1~~RA1 ~~Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.~~ ~~*Op. Modes: All*~~	~~AS1~~RS1 ~~Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.~~ ~~*Op. Modes: All*~~	~~AG1~~RG1 ~~Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.~~ ~~*Op. Modes: All*~~
~~AU2~~RU2 ~~UNPLANNED loss of water level above irradiated fuel.~~ ~~*Op. Modes: All*~~	~~AA2~~RA2 ~~Significant lowering of water level above, or damage to, irradiated fuel.~~ ~~*Op. Modes: All*~~	~~AS2~~RS2 ~~Spent fuel pool level at (site-specific Level 3 description)~~40 ft 8 in (Level 3). ~~*Op. Modes: All*~~	~~AG2~~RG2 ~~Spent fuel pool level cannot be restored to at least (site-specific Level 3 description)~~40 ft 8 in (Level 3) ~~for 60 minutes or longer.~~ ~~*Op. Modes: All*~~
	~~AA3~~RA3 ~~Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.~~ ~~*Op. Modes: All*~~		

~~AU1~~RU1

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: All

Emergency Action Levels:

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

Notes:

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

~~1~~RU1.1 Reading on **ANY** of the following effluent radiation monitors greater than the reading shown for ~~2 times the (site-specific effluent release controlling document) limits for~~ 60 minutes or longer:

Monitor	Reading
1(2)-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation	1.4E-2 µCi/cc
2-RE-305 CTMNT Purge Exhaust Low Range Gas with both purge and GS building ventilation in operation	9.4E-3 µCi/cc
2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation	9.4E-3 µCi/cc
2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only GS building ventilation in operation	2.8E-2 µCi/cc
2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only forced vent of containment	1.0E+1 µCi/cc
2-RE-309 CTMNT Purge Exhaust High Range Gas with only forced vent of containment	1.0E+1 µCi/cc
RE-315 AB Exhaust Low Range Gas	5.4E-3 µCi/cc
RE-317 AB Exhaust Mid-Range Gas	5.4E-3 µCi/cc
RE-325 Drumming Area Exhaust Low Range Gas	8.4E-3 µCi/cc
RE-327 Drumming Area Exhaust Mid-Range Gas	8.4E-3 µCi/cc
1(2)-RE-229 Service Water Overboard	2.3E-3 µCi/cc

RU1.2~~2~~ ~~(site-specific monitor list and threshold values corresponding to 2 times the controlling document limits)~~

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

RU1.3 Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ~~(site-specific effluent release controlling document)~~ODCM limits for 60 minutes or longer.

Definitions:

None

Basis:

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

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

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

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

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

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

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

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

Escalation of the emergency classification level would be via IC ~~AA1~~RA1.

~~**Developer Notes:**~~

~~The "site-specific effluent release controlling document" is the Radiological Effluent Technical Specifications (RETS) or, for plants that have implemented Generic Letter 89-01[1], the~~

~~[1] *Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program*~~

~~Offsite Dose Calculation Manual (ODCM). These documents implement regulations related to effluent controls (e.g., 10 CFR Part 20 and 10 CFR Part 50, Appendix I). As appropriate, the RETS or ODCM methodology should be used for establishing the monitor thresholds for this IC.~~

~~Listed monitors should include the effluent monitors described in the RETS or ODCM.~~

~~Developers may also consider including installed monitors associated with other potential effluent pathways that are not described in the RETS or ODCM[1,2]. If included, EAL values for these monitors should be determined using the most applicable dose/release limits presented in the RETS or ODCM. It is recognized that a calculated EAL value may be below what the monitor can read; in that case, the monitor does not need to be included in the list. Also, some monitors may not be governed by Technical Specifications or other license-related related requirements; therefore, it is important that the associated EAL and basis section clearly identify any limitations on the use or availability of these monitors.~~

~~Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.~~

~~Radiation monitor readings should reflect values that correspond to a radiological release exceeding 2 times a release control limit. The controlling document typically describes methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values. In cases where a methodology is not adequately defined, developers should determine values consistent with effluent control regulations (e.g., 10 CFR Part 20 and 10 CFR Part 50 Appendix I) and related guidance.~~

~~For EAL #2 - Values in this EAL should be 2 times the setpoint established by the radioactivity discharge permit to warn of a release that is not in compliance with the specified limits. Indexing the value in this manner ensures consistency between the EAL and the setpoint established by a specific discharge permit.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within~~

~~[1] This includes consideration of the effluent monitors described in the site emergency plan section(s) which address the requirements of 10 CFR 50.47(b)(8) and (9).~~

~~[2] Developers should keep in mind the requirements of 10 CFR 50.54(q) and the guidance provided by INPO related to emergency response equipment when considering the addition of other effluent monitors.~~

~~the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.1.B~~

~~AU2~~RU2

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

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

- Spent fuel pool low water level alarm
- Visual observation~~(site-specific level indications).~~

AND

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

- ~~(site-specific list of area radiation monitors)~~RE-105 SFP Area Low Range Radiation Monitor
- RE-135 SFP Area High Range Radiation Monitor
- 1(2)-RE-102 El. 66' CONTAINMENT Low Range Monitor

Definitions:

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

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

Basis:

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

A water level decrease will be primarily determined by indications from available level instrumentation. The low level alarm is actuated by LC-634, SFP Level Indicator at 62 ft. 8 in. based on maintaining at least 6 ft. of water on a withdrawn fuel assembly. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

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

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

Escalation of the emergency classification level would be via IC ~~AA2~~RA2.

~~**Developer Notes:**~~

~~The "site-specific level indications" are those indications that may be used to monitor water level in the various portions of the REFUELING PATHWAY. Specify the mode applicability of a particular indication if it is not available in all modes.~~

~~The "site-specific list of area radiation monitors" should contain those area radiation monitors that would be expected to have increased readings following a decrease in water level in the site-specific REFUELING PATHWAY. In cases where a radiation monitor(s) is not available or would not provide a useful indication, consideration should be given to including alternate indications such as UNPLANNED changes in tank and/or sump levels.~~

~~Development of the EALs should consider the availability and limitations of mode-dependent, or other controlled but temporary, radiation monitors. Specify the mode applicability of a particular monitor if it is not available in all modes.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1.B~~

~~AA1~~RA1

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2 or 3 or 4)~~

Notes:

- The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RA1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RA1.1 Reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer:

Monitor	Reading
1(2)-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation	6.0E+0 µCi/cc
1(2)-RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation	6.0E+0 µCi/cc
2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation	4.0E+0 µCi/cc
2-RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation	4.0E+0 µCi/cc
2-RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation	1.2E+1 µCi/cc
RE-317 AB Exhaust Mid-Range Gas	1.0E+0 µCi/cc
RE-319 AB Exhaust High Range Gas	1.0E+0 µCi/cc
RE-327 Drumming Area Exhaust Mid-Range Gas	1.6E+0 µCi/cc

RA1.2 ~~(site-specific monitor list and threshold values)~~

Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond ~~(site-specific dose receptor point)~~SITE BOUNDARY.

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

R~~a~~A1.4 Field survey results indicate **EITHER** of the following at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY:

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

Definitions:

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

Basis:

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

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

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

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

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

For EAL RA1.3, there are no site-specific liquid radiation monitors capable of monitoring liquid effluent releases at the classification threshold for this EAL because their detector operating range is exceeded prior to reaching these levels. Entry into this EAL for a liquid radioactivity release will be based on~~.~~ sampling initiated due to entry into EAL RU1. In practical terms, this means that entry into IC RU1 will start sampling (per RMS Alarm Setpoint and Response Book) which will then allow detection of the setpoint for RA1.

Escalation of the emergency classification level would be via IC ~~AS1~~RS1.

~~**Developer Notes:**~~

~~While this IC may not be met absent challenges to one or more fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Updated Final Safety Analysis Report, the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE….".~~

~~The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous and liquid effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 10 mrem TEDE or 50 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AS1 and AG1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AS1 and AG1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and~~

~~identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~AA2~~RA2

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels:

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

R~~A~~2.1 Uncovery of irradiated fuel in the REFUELING PATHWAY.

R~~A~~2.2 Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by a reading on **ANY** of the following radiation monitors greater than the value shown~~ANY of the following radiation monitors:~~:

Monitor	Reading
RE-105 SFP Area Low Range Radiation Monitor	4 R/hr
1(2)-RE-126 Containment High Radiation Monitor	7 R/hr
1(2)-RE-127 Containment High Radiation Monitor	7 R/hr
1(2)-RE-128 Containment High Radiation Monitor	7 R/hr

RA2.3 ~~(site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)~~

Lowering of spent fuel pool level to ~~(site-specific Level 2 value). [*See Developer Notes*]~~49 -ft.0 in.

Definitions:

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

Basis:

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

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-~~H~~U1.

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

EAL RA2.1

This EAL escalates from ~~AU2~~ RU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used ~~(e.g., a boil-off curve)~~. Classification of an event using this EAL should be based on the totality of available indications, reports, and observations.

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

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

EAL RA2.2

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

EAL RA2.3

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

Escalation of the emergency classification level would be via ICs ~~AS1~~ RS1 or ~~AS2~~ RS2~~(*see AS2 Developer Notes*)~~.

~~**Developer Notes:**~~

~~For EAL #1~~

~~Depending upon the availability and range of instrumentation, this EAL may include specific readings indicative of fuel uncovery; consider water and radiation level readings. Specify the mode applicability of a particular indication if it is not available in all modes.~~

~~For EAL #2~~

~~The "site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms" should contain those radiation monitors that could be used to identify damage to an irradiated fuel assembly (e.g., confirmatory of a release of fission product gases from irradiated fuel).~~

~~For EALs #1 and #2~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL~~

~~values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build in an appropriate level of corroboration between monitor readings into the classification assessment.~~

~~Development of the EALs should also consider the availability and limitations of mode-dependent, or other controlled but temporary, radiation monitors. Specify the mode applicability of a particular monitor if it is not available in all modes.~~

~~For EAL #3~~

~~In accordance with the discussion in Section 1.4, NRC Order EA-12-051, it is recommended that this EAL be implemented when the enhanced spent fuel pool level instrumentation is available for use. The "site-specific Level 2 value" is usually the spent fuel pool level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owner's group guidance.~~

~~Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 2 value.~~

~~ECL Assignment Attributes: 3.1.2.B and 3.1.2.C~~

~~AA3~~RA3

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example 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.

R~~A~~A3.1 Dose rate greater than 15 mR/hr in **ANY** of the following areas:

- Control Room (RE-101)
- Central Alarm Station **AND**
- ~~(other site-specific areas/rooms)~~Secondary Alarm Station (by survey)

R~~A~~A3.2 An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas:

Area	Mode
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B32 MCC Area	4

~~(site-specific list of plant rooms or areas with entry-related mode applicability identified)~~

Definitions:

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

Basis:

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

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

For EAL RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

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

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

The list of plant rooms or areas in EAL RA3.2 was generated from a step-by-step review of OP-3A, 3B, 3C, 5D, and 7A.

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

~~**Developer Notes:**~~

~~EAL #1~~

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

~~The "other site-specific areas/rooms" should include any areas or rooms requiring continuous occupancy to maintain normal plant operation, or to perform a normal cooldown and shutdown.~~

~~EAL #2~~

~~The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed. (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.~~

~~The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~

~~If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.~~

~~Rooms and areas listed in EAL #1 do not need to be included in EAL #2, including the Control Room.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~AS1~~RS1

ECL: Site Area Emergency

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

Operating Mode Applicability: All

Emergency Action Levels:

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

Notes:

- The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RS1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RS1.1 Reading on **ANY** of the following radiation monitor~~s~~ greater than the reading shown for 15 minutes or longer:

Monitor	**Reading**
1(2)-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation	6.0E+1 µCi/cc
1(2)-RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation	6.0E+1 µCi/cc
2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation	4.0E+1 µCi/cc
2-RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation	4.0E+1 µCi/cc
2-RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation	1.2E+2 µCi/cc
RE-317 AB Exhaust Mid-Range Gas	1.0E+1 µCi/cc
RE-319 AB Exhaust High Range Gas	1.0E+1 µCi/cc
RE-327 Drumming Area Exhaust Mid-Range Gas	1.6E+1 µCi/cc

RS1. ~~(site-specific monitor list and threshold values)~~

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

RS1.3 Field survey results indicate **EITHER** of the following at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY:

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

Definitions:

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

Basis:

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

This IC is modified by a note that EAL RS1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions.

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

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

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

Escalation of the emergency classification level would be via IC ~~AG1~~RG1.

~~**Developer Notes:**~~

~~While this IC may not be met absent challenges to multiple fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Updated Final Safety Analysis Report, the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE…."~~

~~The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed~~

~~by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 100 mrem TEDE or 500 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AA1 and AG1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AA1 and AG1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs.~~

~~Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.3.C~~

~~AS2~~RS2
~~*[See Developer Notes]*~~

ECL: Site Area Emergency

Initiating Condition: Spent fuel pool level at ~~(site-specific Level 3 description)~~40 ft. 8 in.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

RS2.1 Lowering of spent fuel pool level to 40 ft. 8 in.~~(site-specific Level 3 value)~~.

Definitions:

None

Basis:

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

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

Escalation of the emergency classification level would be via IC ~~AG1~~ RG1 or ~~AG2~~RG2.

~~**Developer Notes:**~~

~~In accordance with the discussion in Section 1.4, NRC Order EA-12-051, it is recommended that this IC and EAL be implemented when the enhanced spent fuel pool level instrumentation is available for use. The "site-specific Level 3 value" is usually that spent fuel pool level where fuel remains covered and actions to implement make-up water addition should no longer be deferred. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owner's-group guidance.~~

~~Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 3 value.~~

~~ECL Assignment Attributes: 3.1.3.B~~

~~A~~RG1

ECL: General Emergency

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

Operating Mode Applicability: All

Emergency Action Levels:

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

Notes:

- The Emergency Director should declare the General Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RG1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RG1.1 Reading on **ANY** of the following radiation monitor~~s~~ greater than the reading shown for 15 minutes or longer:

Monitor	Reading
1(2)-RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation	6.0E+2 μCi/cc
2-RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation	4.0E+2 μCi/cc
2-RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation	1.2E+3 μCi/cc
RE-317 AB Exhaust Mid-Range Gas	1.0E+2 μCi/cc
RE-319 AB Exhaust High Range Gas	1.0E+2 μCi/cc

~~(site-specific monitor list and threshold values)~~

RG1.2 Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY.

RG1.3 Field survey results indicate **EITHER** of the following at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY:

- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.

- Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.

Definitions:

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

Basis:

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

This IC is modified by a note that EAL RG1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions.

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

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

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

~~**Developer Notes:**~~

~~The effluent ICs/EALs are included to provide a basis for classifying events that cannot be readily classified on the basis of plant conditions alone. The inclusion of both types of ICs/EALs more fully addresses the spectrum of possible events and accidents.~~

~~While this IC may not be met absent challenges to multiple fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Updated Final Safety Analysis Report, the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE….".~~

~~The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed~~

~~by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 1,000 mrem TEDE or 5,000 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AA1 and AS1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AA1 and AS1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within~~

~~the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.4.C~~

~~AG2~~RG2

~~*[See Developer Notes]*~~

ECL: General Emergency

Initiating Condition: Spent fuel pool level cannot be restored to at least 40 ft.8 in. ~~(site-specific Level 3 description)~~ for 60 minutes or longer.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

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

RG2.1 Spent fuel pool level cannot be restored to at least 40 ft. 8 in. ~~(site-specific Level 3 value)~~ for 60 minutes or longer.

Definitions:

None

Basis:

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

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

~~**Developer Notes:**~~

~~In accordance with the discussion in Section 1.4, NRC Order EA-12-051, it is recommended that this IC and EAL be implemented when the enhanced spent fuel pool level instrumentation is available for use. The "site-specific Level 3 value" is usually that spent fuel pool level where fuel remains covered and actions to implement make-up water addition should no longer be deferred. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owner's group guidance.~~

~~Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 3 value.~~

~~ECL Assignment Attributes: 3.1.4.C~~

7 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

~~Table C-1: Recognition Category "C" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~**CU1** UNPLANNED loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory for 15 minutes or longer.~~ ~~*Op. Modes: Cold Shutdown, Refueling*~~*5, 6*	~~**CA1** Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~	~~**CS1** Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory affecting core decay heat removal capability.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~	~~**CG1** Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory affecting fuel clad integrity with containment challenged.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~
~~**CU2** Loss of all but one AC power source to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling, Defueled*~~	~~**CA2** Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling, Defueled*~~		
~~**CU3** UNPLANNED increase in RCS temperature.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~	~~**CA3** Inability to maintain the plant in cold shutdown.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~		
~~**CU4** Loss of Vital DC power for 15 minutes or longer.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~			
~~**CU5** Loss of all onsite or offsite communications capabilities.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold*~~ Shutdown, ~~*Refueling, Defueled*~~			

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
	~~**CA6** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.~~ *~~Op. Modes: 5, 6Cold Shutdown, Refueling~~*		

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

CU1

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: UNPLANNED loss of ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ inventory for 15 minutes or longer.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6

Emergency Action Levels:

~~**Example Emergency Action Levels: (1 or 2)**~~

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

CU1.1 UNPLANNED loss of reactor coolant results ~~(~~in ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level less than a required lower limit for 15 minutes or longer.

CU1.2 a. ~~(~~Reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level cannot be monitored.

AND

b. UNPLANNED increase in ~~(site-specific sump and/or tank)~~Containment Sump A **OR** Waste Holdup Tank levels.

Definitions:

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

Basis:

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

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

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

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

EAL CU1.2 addresses a condition where all means to determine ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~.

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

~~**Developer Notes:**~~

~~EAL #1 – It is recognized that the minimum allowable reactor vessel/RCS/RPV level may have many values over the course of a refueling outage. Developers should solicit input from licensed operators concerning the optimum wording for this EAL statement. In particular, determine if the generic wording is adequate to ensure accurate and timely classification, or if specific setpoints can be included without making the EAL statement unwieldy or potentially inconsistent with actions that may be taken during an outage. If specific setpoints are included, these should be drawn from applicable operating procedures or other controlling documents.~~

~~EAL #2.b – Enter any "site-specific sump and/or tank" levels that could be expected to increase if there were a loss of inventory (i.e., the lost inventory would enter the listed sump or tank).~~

~~ECL Assignment Attributes: 3.1.1.A~~

CU2

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6, Defueled

~~**Example**~~ **Emergency Action Levels:**

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

CU2.1 a. AC power capability to ~~(site-specific emergency buses)~~1(2~~)~~)-A-05 and 1(2)-A-06 is reduced to a single power source for 15 minutes or longer.

AND

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

Definitions:

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

Basis:

Note: with respect to this EAL, "Station Blackout is Unit 1(2) specific."

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.

- Normal Unit 1(2) offsite power sources include:
  - 345 KVAC 1(2)X-03 through the 13.8 KVAC system to the LVSAT 1(2)X-04
  - 345 KVAC backfed through the 19 KVAC system to the UAT 1(2)X-02

- Normal Unit 1(2) onsite power sources consist of:
  - emergency diesel generators
  - gas turbine generator
  - unit main turbine generator
  - power supplied from the opposite unit

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service.

Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

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

An "AC power source" is a source recognized in AOPs and EOPs (including Beyond Design Basis event procedures), and capable of supplying required power to an emergency bus. Some examples of this Initiating ~~condition~~ Condition are presented below.

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

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

~~Developer Notes:~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide required power to an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50%-capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~Developers should modify the bulleted examples provided in the basis section, above, as needed to reflect their site-specific plant designs and capabilities.~~

~~The EALs and Basis should reflect that each independent offsite power circuit constitutes a single power source. For example, three independent 345kV offsite power circuits (i.e., incoming power lines) comprise three separate power sources. Independence may be determined from a review of the site-specific UFSAR, SBO analysis or related loss of electrical power studies.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is recognized in AOPs and EOPS, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.1.A~~

CU3

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: UNPLANNED increase in RCS temperature.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6

Emergency Action Levels:

~~**Example Emergency Action Levels: (1 or 2)**~~

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

CU3.1 UNPLANNED increase in RCS temperature to greater than ~~(site-specific Technical Specification cold shutdown temperature limit)~~200°F.

CU3.2 Loss of **ALL** RCS temperature and ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level indication for 15 minutes or longer.

Definitions:

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

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

Basis:

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

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

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

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

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

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

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

~~Developer Notes:~~

~~For EAL #1, enter the "site-specific Technical Specification cold shutdown temperature limit" where indicated.~~

~~ECL Assignment Attributes: 3.1.1.A~~

CU4

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6

~~Example Emergency~~Emergency Action Levels:

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

CU4.1 Indicated voltage is less than ~~(site-specific bus voltage value)~~115 VDC on required Vital DC buses D-01, D-02, D-03, or D-04 for 15 minutes or longer.

Definitions:

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

Basis:

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

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

The safety-related 125 VDC system consist of four main buses: D-01, D-02, D-03, and D-04.

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

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

~~Developer Notes:~~

~~The "site-specific bus voltage value" should be based on the minimum bus voltage necessary for adequate operation of SAFETY SYSTEM equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate~~

~~those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed.~~

~~The typical value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries, the cell voltage is approximately 1.75 Volts per cell. For a 58 string battery set, the minimum voltage is approximately 1.81 Volts per cell.~~

~~ECL Assignment Attributes: 3.1.1.A~~

CU5

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6, Defueled

Emergency Action Levels:

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

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

- ~~(site-specific list of communications methods)~~Plant Public Address System (Gai-Tronics)
- Commercial Phones
- PBX Phones
- Security Radio
- Portable Radios

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

- Nuclear Accident Reporting System (NARS)
- Commercial Phones
- PBX Phones
- Satellite Phones
- Manitowoc County Sheriff's Department Radio

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

- ~~(site-specific list of communications methods)~~FTS Phone System
- Commercial Phones ~~System~~
- PBX Phones
- Satellite Phones

Definitions:

None

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 ~~ORO~~offsite response organizations and the NRC.

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

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

EAL CU5.2 addresses a total loss of the communications methods used to notify all ~~ORO~~offsite response organizations of an emergency declaration. The offsite response organizations referred to here are the State of Wisconsin, Manitowoc County, and Kewaunee County~~The OROs referred to here are (see Developer Notes)~~.

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

~~**Developer Notes:**~~

~~EAL #1 - The "site-specific list of communications methods" should include all communications methods used for routine plant communications (e.g., commercial or site telephones, page-party systems, radios, etc.). This listing should include installed plant equipment and components, and not items owned and maintained by individuals.~~

~~EAL #2 - The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to OROs as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. Example methods are ring-down/dedicated telephone lines, commercial telephone lines, radios, satellite telephones and internet-based communications technology.~~

~~In the Basis section, insert the site-specific listing of the OROs requiring notification of an emergency declaration from the Control Room in accordance with the site Emergency Plan, and typically within 15 minutes.~~

~~EAL #3 - The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to the NRC as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. These methods are typically the dedicated Emergency Notification System (ENS) telephone line and commercial telephone lines.~~

~~ECL Assignment Attributes: 3.1.1.C~~

CA1

ECL: Alert

Initiating Condition: Loss of ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ inventory.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6

Emergency Action Levels:

~~**Example Emergency**~~~~**Emergency Action Levels:** (1 or 2)~~

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

CA1.1 Loss of ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ inventory as indicated by level less than ~~(site-specific level)~~16% on LI-447 / LI-447A.

CA1.2 a. ~~(~~Reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED increase in ~~(site-specific sump and/or tank)~~Containment Sump A **OR** Waste Holdup Tank levels due to a loss of ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ inventory.

Definitions:

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

Basis:

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

For EAL CA1.1, a lowering of water level below ~~(site-specific level)~~16% on LI-447 / LI-447A indicates that operator actions have not been successful in restoring and maintaining ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery. The LI-447/LI-447A threshold corresponds to the minimum shutdown reactor vessel level required for operation of RHR without air binding the suction.

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

~~For EAL #1 the LI-447/LI-447A threshold corresponds to 6 inches above the bottom inside diameter of the RCS loop. This condition will result in a minimum classification of Alert.~~
For EAL CA1.2, the inability to monitor ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~.

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

If the ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

~~**Developer Notes:**~~

~~For EAL #1 – the "site-specific level" should be based on either:~~

- ~~[*BWR*] Low-Low ECCS actuation setpoint/Level 2. This setpoint was chosen because it is a standard operationally significant setpoint at which some (typically high pressure ECCS) injection systems would automatically start and is a value significantly below the low RPV water level RPS actuation setpoint specified in IC CU1.~~
- ~~[*PWR*] The minimum allowable level that supports operation of normally used decay heat removal systems (e.g., Residual Heat Removal or Shutdown Cooling). If multiple levels exist, specify each along with the appropriate mode or configuration dependency criteria.~~

~~For EAL #2 – The type and range of RCS level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for a PWR. As appropriate to the plant design, alternate means of determining RCS level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. The instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.~~

~~Enter any "site-specific sump and/or tank" levels that could be expected to increase if there were a loss of inventory (i.e., the lost inventory would enter the listed sump or tank).~~

~~ECL Assignment Attributes: 3.1.2.B~~

CA2

ECL: Alert

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

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6, Defueled

Emergency Action Levels:

~~**Example EmergencyEmergency Action Levels:**~~

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

CA2.1 Loss of **ALL** offsite and **ALL** onsite AC Power to ~~(site-specific emergency buses)~~1(2)-A-05 and 1(2)-A-06 for 15 minutes or longer.

Definitions:

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

Basis:

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

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

For the purpose of classification under this EAL, evaluation of power sources should be made on each unit individually.

If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.

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

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

~~Developer Notes:~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide adequate power to an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50%-capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.2.B~~

CA3

ECL: Alert

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

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2)~~

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

~~1~~CA3.1 UNPLANNED increase in RCS temperature to greater than ~~(site-specific Technical Specification cold shutdown temperature limit)~~200°F for greater than the duration specified in the following table.

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

~~2~~CA3.2 UNPLANNED RCS pressure increase greater than ~~(site-specific pressure reading)~~25 psig. (This EAL does not apply during water-solid plant conditions. ~~[*PWR*]~~)

Definitions:

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

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

Basis:

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

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

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact, or RCS inventory is reduced (e.g., mid-loop operation in PWRs). The 20-minute criterion was included to allow time for operator action to address the temperature increase.

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

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact or is at reduced inventory~~[*PWR*]~~, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because

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

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

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

~~**Developer Notes:**~~

~~For EAL #1 – Enter the "site-specific Technical Specification cold shutdown temperature limit" where indicated. The RCS should be considered intact or not intact in accordance with site-specific criteria.~~

~~For EAL #2 – The "site-specific pressure reading" should be the lowest change in pressure that can be accurately determined using installed instrumentation, but not less than 10 psig.~~

~~For PWRs, this IC and its associated EALs address the concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*. A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show that there are sequences that can cause core uncovery in 15 to 20 minutes, and severe core damage within an hour after decay heat removal is lost. The allowed time frames are consistent with the guidance provided by Generic Letter 88-17 and believed to be conservative given that a low pressure Containment barrier to fission product release is established.~~

~~ECL Assignment Attributes: 3.1.2.B~~

CA6

ECL: Alert

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

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6

Emergency Action Levels:

~~**Example Emergency Action Levels:**~~ :

Notes:

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

CA6.1 a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- ~~(site-specific hazards)~~Lake level greater than or equal to 9.0 ft. (Plant elevation)
- Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

AND

b. ~~**EITHER** of the following:~~

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

~~**OR**~~ **AND**

2. ~~2~~**EITHER** of the following:

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

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

Definitions:

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

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

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

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

Basis:

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

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

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

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

~~EAL 1.b.1 addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.~~

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

Escalation of the emergency classification level would be via IC ~~CS1 or AS1~~RS1.

~~**Developer Notes:**~~

~~For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).~~

~~Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.~~

~~ECL Assignment Attributes: 3.1.2.B~~

CS1

ECL: Site Area Emergency

Initiating Condition: Loss of ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ inventory affecting core decay heat removal capability.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6

Emergency Action Levels:

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

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

~~1 a. CONTAINMENT CLOSURE not established.~~

~~**AND**~~

~~b. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level)~~

~~2 a. CONTAINMENT CLOSURE established.~~

~~**AND**~~

~~b. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level).~~

CS1.~~3~~1 a. ~~(~~Reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level cannot be monitored for 30 minutes or longer.

AND

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

- ~~(Site-specific radiation monitor)~~Containment High Radiation Monitor, ~~(~~1(2)-RE-126, 1(2)RE-127, or 1(2)-RE--128,~~)~~ reading greater than ~~(site-specific value)~~100 R/hr
- Erratic source range monitor indication ~~[*PWR*]~~
- UNPLANNED increase in ~~(site-specific sump and/or tank)~~Containment Sump A **OR** Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery
- ~~(Other site-specific indications)~~

Definitions:

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

Basis:

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

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

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

In EAL CS1.~~3~~1.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

For EAL CS1.~~3~~1.a, the calculated radiation level on the Containment High Radiation Monitors (RE-126, RE-127, or RE-128) is without the reactor head in place. Calculated radiation levels with the reactor head in place are below the usable scale of these monitors.

The inability to monitor ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level may be caused by instrumentation and/or power failures, equipment not calibrated for the plant conditions (e.g., hot cal only), or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~.

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

Escalation of the emergency classification level would be via IC CG1 or ~~AG1~~RG1.

~~**Developer Notes:**~~

~~Accident analyses suggest that fuel damage may occur within one hour of uncovery depending upon the amount of time since shutdown; refer to Generic Letter 88-17, SECY 91-283, NUREG-1449 and NUMARC 91-06.~~

~~The type and range of RCS level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for a PWR. As appropriate to the plant design, alternate means of determining RCS level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. The instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.~~

~~PWR~~

~~For EAL #1.b – the "site-specific level" is 6" below the bottom ID of the RCS loop. This is the level at 6" below the bottom ID of the reactor vessel penetration and not the low point of the loop. If the availability of on-scale level indication is such that this level value can be determined during some shutdown modes or conditions, but not others, then specify the mode-dependent and/or configuration states during which the level indication is applicable. If the design and operation of water level instrumentation is such that this level value cannot be determined at any time during Cold Shutdown or Refueling modes, then do not include EAL #1 (classification will be accomplished in accordance with EAL #3).~~

~~For EAL #2.b – The "site-specific level" should be approximately the top of active fuel. If the availability of on-scale level indication is such that this level value can be determined during some shutdown modes or conditions, but not others, then specify the mode-dependent and/or configuration states during which the level indication is applicable. If the design and operation of water level instrumentation is such that this level value cannot be determined at any time during Cold Shutdown or Refueling modes, then do not include EAL #2 (classification will be accomplished in accordance with EAL #3).~~

~~For EAL #3.b – first bullet – As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site-specific value" indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.~~

~~For EAL #3.b – second bullet – Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.~~

~~For EAL #3.b – third bullet – Enter any 'site-specific sump and/or tank" levels that could be expected to change if there were a loss of RCS/reactor vessel inventory of sufficient magnitude to indicate core uncovery. Specific level values may be included if desired.~~

~~For EAL #3.b – fourth bullet - Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.~~

~~BWR~~

~~For EAL #1.b – "site-specific level" is the Low-Low-Low ECCS actuation setpoint / Level 1. The BWR Low-Low-Low ECCS actuation setpoint / Level 1 was chosen because it is a standard operationally significant setpoint at which some (typically low pressure ECCS) injection systems would automatically start and attempt to restore RPV level. This is a RPV water level value that is observable below the Low-Low/Level 2 value specified in IC CA1, but significantly above the Top of Active Fuel (TOAF) threshold specified in EAL #2.~~

~~For EAL #2.b – The "site-specific level" should be for the top of active fuel.~~

~~For EAL #3.b – first bullet - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site-specific value" indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.~~

~~For BWRs that do not have installed radiation monitors capable of indicating core uncovery, alternate site-specific level indications of core uncovery should be used if available.~~

~~For EAL #3.b – second bullet - Because BWR source range monitor (SRM) nuclear instrumentation detectors are typically located below core mid-plane, this may not be a viable indicator of core uncovery for BWRs.~~

~~For EAL #3.b – third bullet – Enter any "site-specific sump and/or tank" levels that could be expected to change if there were a loss of RPV inventory of sufficient magnitude to indicate core uncovery. Specific level values may be included if desired.~~

~~For EAL #3.b – fourth bullet - Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.~~

~~ECL Assignment Attributes: 3.1.3.B~~

CG1

ECL: General Emergency

Initiating Condition: Loss of ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ inventory affecting fuel clad integrity with containment challenged.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~5, 6

Emergency Action Levels:

~~**Example Emergency**~~**Emergency** ~~**Action Levels:** (1 or 2)~~

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

~~1 a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level) for 30 minutes or longer.~~

~~**AND**~~

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

CG1.~~2~~1

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

AND

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

- ~~(Site-specific radiation monitor)~~Containment High Radiation Monitors, 1(2)RE-126, 1(2)RE-127, or 1(2)RE-128, ~~(1(2)-RE-126, RE-127, or RE-128)~~ reading greater than 100 R/hr~~Containment High Range Monitor reading greater than (site-specific value) R/hr~~
- Erratic source range monitor indication ~~[*PWR*]~~
- UNPLANNED increase in ~~(site-specific sump and/or tank)~~Containment Sump A **OR** Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery
- ~~(Other site-specific indications)~~

AND

c. **ANY** indication from the Containment Challenge Table ~~(see below)~~C-1.

Containment Challenge Table C-1
■ CONTAINMENT CLOSURE not established*
■ ~~(Explosive mixture)~~6% H^2 exists inside containment
■ UNPLANNED increase in containment pressure
■ ~~Secondary containment radiation monitor reading above (site-specific value) [*BWR*]~~

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

Definitions:

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

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

Basis:

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

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

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

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

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

In EAL CG1.~~2~~1.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

For EAL CG1.~~2~~1.b, the calculated radiation level on the Containment High Radiation Monitors,

1(2)RE-126, 1(2)RE-127, or 1(2)RE-128. ~~(1(2) RE-126, RE-127, or RE-128)~~ is without the reactor head in place. Calculated radiation levels with the reactor head in place are below the usable scale of these monitors.

The inability to monitor ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~ level may be caused by instrumentation and/or power failures, equipment not calibrated for the plant conditions (e.g., hot cal only).~~,~~ or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(~~reactor vessel/RCS ~~[*PWR*] or RPV [*BWR*])~~.

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

~~**Developer Notes:**~~

~~Accident analyses suggest that fuel damage may occur within one hour of uncovery depending upon the amount of time since shutdown; refer to Generic Letter 88-17, SECY 91-283, NUREG-1449 and NUMARC 91-06.~~

~~The type and range of RCS level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for a PWR. As appropriate to the plant design, alternate means of determining RCS level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. The instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.~~

~~For EAL #1.a – The "site-specific level" should be approximately the top of active fuel. If the availability of on-scale level indication is such that this level value can be determined during some shutdown modes or conditions, but not others, then specify the mode-dependent and/or configuration states during which the level indication is applicable. If the design and operation of water level instrumentation is such that this level value cannot be determined at any time during Cold Shutdown or Refueling modes, then do not include EAL #1 (classification will be accomplished in accordance with EAL #2).~~

~~For EAL #2.b – first bullet – As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site-specific value" indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.~~

~~For BWRs that do not have installed radiation monitors capable of indicating core uncovery, alternate site-specific level indications of core uncovery should be used if available.~~

~~For EAL #2.b – second bullet - Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations. Because BWR Source Range Monitor (SRM) nuclear instrumentation detectors are typically located below core mid-plane, this may not be a viable indicator of core uncovery for BWRs.~~

~~For EAL #2.b – third bullet - Enter any "site-specific sump and/or tank" levels that could be expected to change if there were a loss of inventory of sufficient magnitude to indicate core uncovery. Specific level values may be included if desired.~~

~~For EAL #2.b – fourth bullet - Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.~~

~~For the Containment Challenge Table:~~

~~Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of RCS heat removal or inventory control functions.~~

~~For "Explosive mixture", developers may enter the minimum containment atmospheric hydrogen concentration necessary to support a hydrogen burn (i.e., the lower deflagration limit). A concurrent containment oxygen concentration may be included if the plant has this indication available in the Control Room.~~

~~For BWRs, the use of secondary containment radiation monitors should provide indication of increased release that may be indicative of a challenge to secondary containment. The "site-specific value" should be based on the EOP maximum safe values because these values are easily recognizable and have a defined basis.~~

~~ECL Assignment Attributes: 3.1.4.B~~

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

~~**Table E-1: Recognition Category "E" Initiating Condition Matrix**~~

~~**UNUSUAL EVENT**~~

~~**E-HU1** Damage to a loaded cask CONFINEMENT BOUNDARY.~~
~~*Op. Modes: All*~~

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

ISFSI MALFUNCTION

E-~~H~~U1

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

EU1.1 Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than the values shown below~~(2 times the site-specific cask specific technical specification allowable radiation level)~~ on the surface of the spent fuel cask.

32 PT DSC	
Front Surface	1700 mrem/-hr~~.~~
Door Centerline	400 mrem/-hr~~.~~
End Shield Wall Exterior	1~~6~~2 mrem/-hr~~.~~
VSC-24	
Sides	200 mrem/-hr~~.~~
Top	400 mrem/-hr~~.~~
Air Inlets	700 mrem/-hr~~.~~
Air Outlets	200 mrem/-hr~~.~~

Definition:

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

Basis:

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

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category ~~A~~ R IC ~~AU1~~RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

ISFSI MALFUNCTION

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

~~**Developer Notes:**~~

~~The results of the ISFSI Safety Analysis Report (SAR) [per NUREG 1536], or a SAR referenced in the cask Certificate of Compliance and the related NRC Safety Evaluation Report, identify the natural phenomena events and accident conditions that could potentially affect the CONFINEMENT BOUNDARY. This EAL addresses damage that could result from the range of identified natural or man-made events (e.g., a dropped or tipped over cask, EXPLOSION, FIRE, EARTHQUKE, etc.).~~

~~The allowable radiation level for a spent fuel cask can be found in the cask's technical specification located in the Certificate of Compliance.~~

~~ECL Assignment Attributes: 3.1.1.B~~

9 FISSION PRODUCT BARRIER ICS/EALS

~~Table 9-F-1: Recognition Category "F" Initiating Condition Matrix~~

~~**ALERT**~~	
~~**FA1**~~	~~Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.~~ ~~*Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown1, 2, 3, 4*~~
~~**SITE AREA EMERGENCY**~~	
~~**FS1**~~	~~Loss or Potential Loss of any two barriers.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Hot Standby, Startup, Hot Shutdown*~~
~~**GENERAL EMERGENCY**~~	
~~**FG1**~~	~~Loss of any two barriers and Loss or Potential Loss of the third barrier.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Hot Standby, Startup, Hot Shutdown*~~

~~**See Table 9-F-2 for BWR EALs**~~
~~**See Table 9-F-3 for PWR EALs**~~

~~**Developer Note:** The adjacent logic flow diagram is for use by developers and is not required for site-specific implementation; however, a site-specific scheme must include some type of user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. Such aids are typically comprised of logic flow diagrams, "scoring" criteria or checkbox-type matrices. The user-aid logic must be consistent with that of the adjacent diagram.~~

~~Developer Notes~~

1. ~~The logic used for these initiating conditions reflects the following considerations:~~

   - ~~The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.~~

   - ~~Unusual Event ICs associated with fission product barriers are addressed in Recognition Category S.~~

2. ~~For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC AG1 has been exceeded.~~

3. ~~The fission product barrier thresholds specified within a scheme are expected to reflect plant-specific design and operating characteristics. This may require that developers create different thresholds than those provided in the generic guidance.~~

4. ~~Alternative presentation methods for the Recognition Category F ICs and fission product barrier thresholds are acceptable and include flow charts, block diagrams, and checklist-type tables. Developers must ensure that the site-specific method addresses all possible threshold combinations and classification outcomes shown in the BWR or PWR EAL fission product barrier tables. The NRC staff considers the presentation method of the Recognition Category F information to be an important user aid and may request a change to a particular proposed method if, among other reasons, the change is necessary to promote consistency across the industry.~~

5. ~~As used in this Recognition Category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location – inside containment, a secondary-side system (i.e., PWR steam generator tube leakage), an interfacing system, or outside of containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.~~

6. ~~At the Site Area Emergency level, classification decision-makers should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.~~

7. ~~The ability to escalate to a higher emergency classification level in response to degrading conditions should be maintained. For example, a steady increase in RCS leakage would represent an increasing risk to public health and safety.~~

~~Table 9-F-2: BWR EAL Fission Product Barrier Table~~

~~Thresholds for LOSS or POTENTIAL LOSS of Barriers~~

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

~~Fuel Clad Barrier~~		~~RCS Barrier~~		~~Containment Barrier~~	
~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~
~~1. RCS Activity~~		**~~1. Primary Containment Pressure~~**		**~~1. Primary Containment Conditions~~**	
~~A. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).~~	~~Not Applicable~~	~~A. Primary containment pressure greater than (site-specific value) due to RCS leakage.~~	~~Not Applicable~~	~~A. UNPLANNED rapid drop in primary containment pressure following primary containment pressure rise~~ **~~OR~~** ~~B. Primary containment pressure response not consistent with LOCA conditions.~~	~~A. Primary containment pressure greater than (site-specific value)~~ **~~OR~~** ~~B. (site-specific explosive mixture) exists inside primary containment~~ **~~OR~~** ~~C. HCTL exceeded.~~
~~2. RPV Water Level~~		**~~2. RPV Water Level~~**		**~~2. RPV Water Level~~**	
~~A. Primary containment flooding required.~~	~~A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to~~	~~A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to~~	~~Not Applicable~~	~~Not Applicable~~	~~A. Primary containment flooding required.~~

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
	the top of active fuel) or cannot be determined.	the top of active fuel) or cannot be determined.			
3. Not Applicable		**3. RCS Leak Rate**		**3. Primary Containment Isolation Failure**	
Not Applicable	Not Applicable	A. UNISOLABLE break in **ANY** of the following: (site-specific systems with potential for high-energy line breaks) **OR** B. Emergency RPV Depressurization.	A. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following: 1. Max Normal Operating Temperature **OR** 2. Max Normal Operating Area Radiation Level.	A. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal **OR** B. Intentional primary containment venting per EOPs **OR** C. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following: 1. Max Safe Operating Temperature. **OR**	Not Applicable

~~Fuel Clad Barrier~~		~~RCS Barrier~~		~~Containment Barrier~~	
~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~
				~~2. Max Safe Operating Area Radiation Level.~~	
~~4. Primary Containment Radiation~~		~~4. Primary Containment Radiation~~		~~4. Primary Containment Radiation~~	
~~A. Primary containment radiation monitor reading greater than (site-specific value).~~	~~Not Applicable~~	~~A. Primary containment radiation monitor reading greater than (site-specific value).~~	~~Not Applicable~~	~~Not Applicable~~	~~A. Primary containment radiation monitor reading greater than (site-specific value).~~
~~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)~~
~~6. Emergency Director Judgment~~		~~6. Emergency Director Judgment~~		~~6. Emergency Director Judgment~~	
~~A. ANY condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.~~	~~A. ANY condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.~~	~~A. ANY condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.~~	~~A. ANY condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.~~	~~A. ANY condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.~~	~~A. ANY condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.~~

~~Basis Information For~~
~~BWR EAL Fission Product Barrier Table 9-F-2~~

~~BWR FUEL CLAD BARRIER THRESHOLDS:~~

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

~~1. RCS Activity~~

~~Loss 1.A~~

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

~~There is no Potential Loss threshold associated with RCS Activity.~~

~~Developer Notes:~~

~~Threshold values should be determined assuming RCS radioactivity concentration equals 300 μCi/gm dose equivalent I-131. Other site-specific units may be used (e.g., μCi/cc).~~

~~Depending upon site-specific capabilities, this threshold may have a sample analysis component and/or a radiation monitor reading component.~~

~~Add this paragraph (or similar wording) to the Basis if the threshold includes a sample analysis component, "It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications."~~

~~2. RPV Water Level~~

~~Loss 2.A~~

~~The Loss threshold represents the EOP requirement for primary containment flooding. This is identified in the BWROG EPGs/SAGs when the phrase, "Primary Containment Flooding Is Required," appears. Since a site-specific RPV water level is not specified here, the Loss threshold phrase, "Primary containment flooding required," also accommodates the EOP need to flood the primary containment when RPV water level cannot be determined and core damage due to inadequate core cooling is believed to be occurring.~~

~~Potential Loss 2.A~~

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

~~BWR FUEL CLAD BARRIER THRESHOLDS:~~

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

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

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

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

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

~~BWR FUEL CLAD BARRIER THRESHOLDS:~~

~~Developer Notes:~~

~~Loss 2.A~~

~~The phrase, "Primary containment flooding required," should be modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs (e.g., drywell flooding required, etc.).~~

~~Potential Loss 2.A~~

~~The decision that "RPV water level cannot be determined" is directed by guidance given in the RPV water level control sections of the EOPs.~~

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

~~4. Primary Containment Radiation~~

~~Loss 4.A~~

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

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

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

~~Developer Notes:~~

~~The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS radioactivity concentration equal to 300 μCi/gm dose equivalent I-131, into the primary containment atmosphere.~~

~~BWR FUEL CLAD BARRIER THRESHOLDS:~~

~~5. Other Indications~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Fuel Clad barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~Developer Notes:~~

~~Loss and/or Potential Loss 5.A~~

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

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

~~6. Emergency Director Judgment~~

~~Loss 6.A~~

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

~~Potential Loss 6.A~~

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

~~Developer Notes:~~

~~None~~

~~BWR RCS BARRIER THRESHOLDS:~~

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

~~1. Primary Containment Pressure~~

~~Loss 1.A~~

~~The (site-specific value) primary containment pressure is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.~~

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

~~Developer Notes:~~

~~None~~

~~2. RPV Water Level~~

~~Loss 2.A~~

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

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

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

~~BWR RCS BARRIER THRESHOLDS:~~

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

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

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

~~3. RCS Leak Rate~~

~~Loss Threshold 3.A~~

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

~~Loss Threshold 3.B~~

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

~~Potential Loss Threshold 3.A~~

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

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

~~BWR RCS BARRIER THRESHOLDS:~~

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

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

~~Developer Notes:~~

~~Loss Threshold 3.A~~

~~The list of systems included in this threshold should be the high energy lines which, if ruptured and remain unisolated, can rapidly depressurize the RPV. These lines are typically isolated by actuation of the Leak Detection system.~~

~~Large high-energy line breaks such as Main Steam Line (MSL), High Pressure Coolant Injection (HPCI), Feedwater, Reactor Water Cleanup (RWCU), Isolation Condenser (IC) or Reactor Core Isolation Cooling (RCIC) that are UNISOLABLE represent a significant loss of the RCS barrier.~~

~~4. Primary Containment Radiation~~

~~Loss 4.A~~

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

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

~~Developer Notes:~~

~~The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS activity at Technical Specification allowable limits, into the primary containment atmosphere. Using RCS activity at Technical Specification allowable limits aligns this threshold with IC SU3. Also, RCS activity at this level will typically result in primary containment radiation levels that can be more readily detected by primary containment radiation monitors, and more readily differentiated from those caused by piping or component "shine" sources. If desired, a plant may use a lesser value of RCS activity for determining this value.~~

~~BWR RCS BARRIER THRESHOLDS:~~

~~In some cases, the site-specific physical location and sensitivity of the primary containment radiation monitor(s) may be such that radiation from a cloud of released RCS gases cannot be distinguished from radiation emanating from piping and components containing elevated reactor coolant activity. If so, refer to the Developer Guidance for Loss/Potential Loss 5.A and determine if an alternate indication is available.~~

~~5. Other Indications~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the RCS barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~Developer Notes:~~

~~Loss and/or Potential Loss 5.A~~

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

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

~~6. Emergency Director Judgment~~

~~Loss 6.A~~

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

~~Potential Loss 6.A~~

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

~~Developer Notes:~~

~~None~~

~~BWR CONTAINMENT BARRIER THRESHOLDS:~~

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

~~1. Primary Containment Conditions~~

~~Loss 1.A and 1.B~~

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

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

~~Potential Loss 1.A~~

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

~~Potential Loss 1.B~~

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

~~Potential Loss 1.C~~

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

- ~~Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,~~

~~OR~~

~~BWR CONTAINMENT BARRIER THRESHOLDS:~~

- ~~Suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.~~

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

~~Developer Notes:~~

~~Potential Loss 1.B~~

~~BWR EPGs/SAGs specifically define the limits associated with explosive mixtures in terms of deflagration concentrations of hydrogen and oxygen. For Mk I/II containments the deflagration limits are "6% hydrogen and 5% oxygen in the drywell or suppression chamber". For Mk III containments, the limit is the "Hydrogen Deflagration Overpressure Limit". The threshold term "explosive mixture" is synonymous with the EPG/SAG "deflagration limits".~~

~~Potential Loss 1.C~~

~~Since the HCTL is defined assuming a range of suppression pool water levels as low as the elevation of the downcomer openings in Mk I/II containments, or 2 feet above the elevation of the horizontal vents in a Mk III containment, it is unnecessary to consider separate Containment barrier Loss or Potential Loss thresholds for abnormal suppression pool water level conditions. If desired, developers may include a separate Containment Potential Loss threshold based on the inability to maintain suppression pool water level above the downcomer openings in Mk I/II containments, or 2 feet above the elevation of the horizontal vents in a Mk III containment with RPV pressure above the minimum decay heat removal pressure, if it will simplify the assessment of the suppression pool level component of the HCTL.~~

~~2. RPV Water Level~~

~~There is no Loss threshold associated with RPV Water Level.~~

~~Potential Loss 2.A~~

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

~~BWR CONTAINMENT BARRIER THRESHOLDS:~~

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

~~Developer Notes:~~

~~The phrase, "Primary containment flooding required," should be modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs (e.g., drywell flooding required, etc.).~~

~~3. Primary Containment Isolation Failure~~

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

~~Loss 3.A~~

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

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

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

~~Loss 3.B~~

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

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

~~Loss 3.C~~

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

~~BWR CONTAINMENT BARRIER THRESHOLDS:~~

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

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

~~There is no Potential Loss threshold associated with Primary Containment Isolation Failure.~~

~~Developer Notes:~~

~~Loss 3.B~~

~~Consideration may be given to specifying the specific procedural step within the Primary Containment Control EOP that defines intentional venting of the Primary Containment regardless of offsite radioactivity release rate.~~

~~4. Primary Containment Radiation~~

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

~~Potential Loss 4.A~~

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

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

~~**Developer Notes:**~~

~~NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, provides the basis for using the 20% fuel cladding failure value. Unless there is a site-specific analysis justifying a different value, the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad failure into the primary containment atmosphere.~~

~~**BWR CONTAINMENT BARRIER THRESHOLDS:**~~

~~**5. Other Indications**~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Containment barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~**Developer Notes:**~~

~~Loss and/or Potential Loss 5.A~~

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

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

~~**6. Emergency Director Judgment**~~

~~Loss 6.A~~

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

~~Potential Loss 6.A~~

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

~~**Developer Notes:**~~

~~None~~

Table 9-F-21~~3~~: ~~PWR~~ EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

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

Fuel Clad Barrier		**RCS Barrier**		**Containment Barrier**	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
A. Conditions requiring entry into Core Cooling RED Path (CSP C.1) are met.	A. Conditions requiring entry into Core Cooling ORANGE Path (CSP C.2) are met. **OR** B. Conditions requiring entry into Heat Sink RED Path (CSP H.1) are met.	Not Applicable	A. Conditions requiring entry into Heat Sink RED Path (CSP H.1) are met. **OR** B. Conditions requiring entry into RCS Integrity RED Path (CSP P.1) are met.	Not Applicable	A. 1. Conditions requiring entry~~Entry~~ into Core Cooling RED Path (CSP C.1) are met. **AND** 2. CSP C.1 not effective within 15 minutes.
~~1~~2. RCS or SG Tube Leakage		**~~1~~2. RCS or SG Tube Leakage**		**~~1~~2. RCS or SG Tube Leakage**	
Not Applicable	Not Applicable ~~A. RCS/reactor vessel level less than (site-specific level).Conditions requiring entry into Core Cooling ORANGE Path (CSP C.2) are met.~~	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** ~~2.~~ SG tube RUPTURE. ~~3.~~	A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** ~~2.~~ SG tube leakage.	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
		~~1.~~2.	2. ~~OR~~ ~~B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).~~~~Conditions requiring entry into RCS Integrity RED Path (CSP-P.1) are met.~~		

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
~~2. Inadequate Heat Removal~~		~~2. Inadequate Heat Removal~~		~~2. Inadequate Heat Removal~~	
~~A. Core exit thermocouple readings greater than (site-specific temperature value).Conditions requiring entry into Core Cooling RED Path (CSP C.1) are met.~~	~~A. Core exit thermocouple readings greater than (site-specific temperature value).Conditions requiring entry into Core Cooling ORANGE Path (CSP C.2) are met.~~ ~~OR~~ ~~B. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).Conditions requiring entry into Heat Sink RED Path (CSP H.1) are met.~~	~~Not Applicable~~	~~A. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).Conditions requiring entry into Heat Sink RED Path (CSP H.1) are met.~~	~~Not Applicable~~	~~A. 1. (Site-specific criteria for entry into core cooling restoration procedure)Entry into Core Cooling RED Path (CSP C.1) are met.~~ ~~AND~~ ~~2. Restoration procedure not effective within 15 minutes.CSP C.1 not effective within 15 minutes.~~

~~3~~3. RCS Activity / Containment Radiation		3~~3~~. RCS Activity / Containment Radiation		~~3~~3. RCS Activity / Containment Radiation	
A. Containment radiation monitor reading greater than 577 R/hr indicated on **ANY** of the following. • 1(2)–RE–126 • 1(2)–RE–127 • 1(2)–RE–128 **OR** B. 1(2)~~- (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).~~RE-109 greater than 4,500 mR/hr ~~.~~	Not Applicable	A. Containment radiation monitor reading greater than 11 R/hr indicated on **ANY** of the following~~.~~: • 1(2~~)~~)–RE–126 • 1(2)–RE–127 • 1(2)–RE–128	Not Applicable	Not Applicable	A. Containment radiation monitor reading greater than 18,500 R/hr indicated on **ANY** of the following. • 1(2)–RE–126 • 1(2)–RE–127 • 1(2)–RE–128 ~~(site-specific value).~~

44. Containment Integrity or Bypass		44. Containment Integrity or Bypass		44. Containment Integrity or Bypass	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Containment isolation is required **AND** **EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment pressure greater than ~~(site-specific value)~~60 psig. **OR** B. ~~Explosive mixture exists inside containment~~6% H^2 inside containment. **OR** C. 1. Containment pressure greater than ~~(site-specific pressure setpoint)~~25 psig. **AND** 2. Less than one full train of ~~(site-specific system or equipment)~~depressurization equipment is operating per design for 15 minutes or longer.
~~**5. Other Indications**~~		~~**5. Other Indications**~~		~~**5. Other Indications**~~	
~~Not ApplicableA. (site-specific as applicable)~~	~~Not ApplicableA. (site-specific as applicable)~~	~~Not ApplicableA. (site-specific as applicable)~~	~~Not ApplicableA. (site-specific as applicable)~~	~~Not ApplicableA. (site-specific as applicable)~~	~~Not ApplicableA. (site-specific as applicable)~~

~~6~~5. Emergency Director Judgment		**5~~6~~. Emergency Director Judgment**		**5~~6~~. Emergency Director Judgment**	
A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis Information For
~~PWR EAL~~ Fission Product Barrier Table 9-F-1

~~**Developer Notes:**~~

~~**Threshold Parameters and Values**~~

~~Each PWR owner's group has developed a methodology for guiding the development and implementation of EOPs (i.e., assessing plant parameters, and determining and prioritizing operator actions). Many of the thresholds contained in the PWR EAL Fission Product Barrier Table reflect conditions that are specifically addressed in EOPs (e.g., a loss of heat removal capability by the steam generators). When developing a site-specific threshold, developers should use the parameters and values specified within their EOPs that align with the condition described by the generic threshold and basis, and related developer notes. This approach will ensure consistency between the site-specific EOPs and emergency classification scheme, and thus facilitate more timely and accurate classification assessments.~~

~~In support of EOP development and implementation, the Westinghouse Owners Group (WOG) developed a defined set of Critical Safety Functions as part of their Emergency Response Guidelines. The WOG approach structures EOPs to maintain and/or restore these Critical Safety Functions, and to do so in a prioritized and systematic manner. The WOG Critical Safety Functions are presented below.~~

- ~~Subcriticality~~
- ~~Core Cooling~~
- ~~Heat Sink~~
- ~~RCS Integrity~~
- ~~Containment~~
- ~~RCS Inventory~~

~~The WOG ERGs provide a methodology for monitoring the status of the Critical Safety Functions and classifying the significance of a challenge to a function; this methodology is referred to as the Critical Safety Function Status Trees (CSFSTs). For plants that have implemented the WOG ERGs, the guidance in NEI 99-01 allows for use of certain CSFST assessment results as EALs and fission product barrier loss/potential loss thresholds. In this manner, an emergency classification assessment may flow directly from a CSFST assessment.~~

~~It is important to understand that the CSFSTs are evaluated using plant parameters, and that they are simply a vendor-specific method for collectively evaluating a set of parameters for purposes of driving emergency operating procedure usage. For the emergency conditions of interest, the generic thresholds within the PWR EAL Fission Product Barrier Table specify the plant parameters that define a potential loss or loss of a fission product barrier; however, as described in the associated Developer Notes, a CSFST terminus may be used as well. For this reason, inclusion of the CSFST-related thresholds would be redundant to the parameter-based thresholds for plants that employ the WOG ERGs.~~

~~Sites that employ the WOG ERGs may, at their discretion, include the CSFST-based loss and potential loss thresholds as described in the Developer Notes. Developers at these sites should consult with their classification decision-makers to determine if inclusion would assist with timely and accurate emergency classification. This decision should consider the effects of any site-specific changes to the generic WOG CSFST evaluation logic and setpoints, as well as those~~

~~arising from user rules applicable to emergency operating procedures (e.g., exceptions to procedure entry or transition due to specific accident conditions or loss of a support system).~~

~~The CSFST thresholds may be addressed in one of 3 ways:~~

~~1) Not incorporated; thresholds will use parameters and values as discussed in the Developer Notes.~~

~~2) Incorporated along with parameter and value thresholds (e.g., a fuel clad loss would have 2 thresholds such as "CETs > 1200°F" and "Core Cooling Red entry conditions met".~~

~~3) Used in lieu of parameters and values for all thresholds.~~

~~With one exception, if a decision is made to include the CSFST-based thresholds, then all such allowed thresholds must be used in the table (e.g., it is not permissible to use only the C Orange terminus as a potential loss of the fuel clad barrier threshold and disregard all other CSFST-based thresholds). The one exception is the RCS Integrity (P) CSFST. Because of the complexity of the P Red decision-point that relies on an assessment a pressure-temperature curve, a P Red condition may be used as an RCS potential loss threshold without the need to incorporate the other CSFST-based thresholds.~~

~~PWR~~ FUEL CLAD BARRIER THRESHOLDS:

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

1. Critical Safety Function Status

Loss 1.A

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

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier. CSP-C.1 is the Critical Safety Procedure that provides directions to restore core cooling.

Potential Loss 1.A

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

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur. CSP-C.2 is the Critical Safety Procedure that provides directions to restore adequate core cooling.

Potential Loss 1.B

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. Verification of the EALs for CSP-H.1 should include an assessment of if feed flow was reduced based on the actions performed for an uncontrolled depressurization of both steam generators. If this is the case, then declaration requirements should not be considered to be met. This does not affect the time that the CSFST initially changed color and met the CSP entry conditions for potential EAL event classification. Alternatively, if CSP-H.1 was entered during a loss of coolant accident, it may be exited if secondary heat sink is not required based on RCS pressure less than non-faulted S/G pressure. If this is the case, then declaration requirements should not be considered to be met.

~~1. RCS or SG Tube Leakage~~

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

~~Potential Loss 1.A~~

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

~~Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur. CSP-C.2 is the Critical Safety Procedure that provides directions to restore adequate core cooling.~~

~~**Developer Notes:**~~
~~Potential Loss 1.A~~

~~Enter the site-specific reactor vessel water level value(s) used by EOPs to identify a degraded core cooling condition (e.g., requires prompt restoration action). The reactor vessel level that corresponds to approximately the top of active fuel may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the reactor vessel level(s) used for the Core Cooling Orange Path (including dependencies upon the status of RCPs, if applicable).~~

~~Westinghouse ERG Plants~~

~~Developers should consider including a threshold the same as, or similar to, "Core Cooling Orange entry conditions met" in accordance with the guidance at the front of this section. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.~~

~~**PWR FUEL CLAD BARRIER THRESHOLDS:**~~

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

In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using this threshold is not warranted.

FUEL CLAD BARRIER THRESHOLDS:

2. RCS or SG Tube Leakage

There are no Loss or Potential Loss thresholds associated with RCS or SG Tube Leakage.

~~Potential Loss 1.A~~

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

~~**Developer Notes:**~~
~~Potential Loss 1.A~~

~~Enter the site-specific reactor vessel water level value(s) used by EOPs to identify a degraded core cooling condition (e.g., requires prompt restoration action). The reactor vessel level that corresponds to approximately the top of active fuel may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the reactor vessel level(s) used for the Core Cooling Orange Path (including dependencies upon the status of RCPs, if applicable).~~

~~Westinghouse ERG Plants~~

~~Developers should consider including a threshold the same as, or similar to, "Core Cooling Orange entry conditions met" in accordance with the guidance at the front of this section. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.~~

~~**Developer Notes:**~~

~~Some site-specific EOPs and/or EOP user guidelines may establish decision-making criteria concerning the number or other attributes of thermocouple readings necessary to drive actions (e.g., 5 CETs reading greater than 1,200°F is required before transitioning to an inadequate core cooling procedure). To maintain consistency with EOPs, these decision-making criteria may be used in the core exit thermocouple reading thresholds.~~

~~Loss 2.A~~

~~Enter a site-specific temperature value that corresponds to significant in-core superheating of reactor coolant. 1,200°F may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Red Path.~~

~~Potential Loss 2.A~~

~~Enter a site-specific temperature value that corresponds to core conditions at the onset of heat-induced cladding damage (e.g., the temperature allowing for the formation of superheated steam assuming that the RCS is intact). 700°F may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Orange Path.~~

~~Potential Loss 2.B~~

~~Enter the site-specific parameters and values that define an extreme challenge to the ability to remove heat from the RCS via the steam generators. These will typically be parameters and values that would require operators to take prompt action to address this condition.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Heat Sink Red Path.~~

~~Westinghouse ERG Plants~~

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

~~**PWR FUEL CLAD BARRIER THRESHOLDS:**~~

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

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

~~**FUEL CLAD BARRIER THRESHOLDS:**~~

~~2.~~3. RCS Activity / Containment Radiation

Loss 3.A

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

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

Loss 3.B

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

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

~~**Developer Notes:**~~

~~Loss 3.A~~

~~The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS radioactivity concentration equal to 300 μCi/gm dose equivalent I-131, into the containment atmosphere.~~

~~**PWR FUEL CLAD BARRIER THRESHOLDS:**~~

~~Loss 3.B~~

~~Threshold values should be determined assuming RCS radioactivity concentration equals 300 μCi/gm dose equivalent I-131. Other site-specific units may be used (e.g., μCi/cc).~~

~~Depending upon site-specific capabilities, this threshold may have a sample analysis component and/or a radiation monitor reading component.~~

~~Add this paragraph (or similar wording) to the Basis if the threshold includes a sample analysis component, "It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications."~~

~~3.~~4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

~~**4. Other Indications**~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Fuel Clad barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~**Developer Notes:**~~

~~Loss and/or Potential Loss 5.A~~

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

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

5. Emergency Director Judgment

Loss 5~~6~~.A

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

~~PWR FUEL CLAD BARRIER THRESHOLDS:~~

Potential Loss 5~~6~~.A

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

~~Developer Notes:~~

~~None~~

~~PWR~~ RCS BARRIER THRESHOLDS:

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

1. Critical Safety Function Status

There is no Loss threshold associated with Critical Safety Function Status.

Potential Loss 1.A

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. If CSP-H.1 was entered during a loss of coolant accident, it may be exited if secondary heat sink is not required based on RCS pressure less than non-faulted S/G pressure. If this is the case, then declaration requirements should not be considered to be met.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold 1.B; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

CSP-H.1 is the Critical Safety Procedure that provides directions if the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a Potential Loss of the RCS barrier.

In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using this threshold is not warranted.

Potential Loss 1.B

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

CSP-P.1 is the Critical Safety Procedure that provides directions to avoid, or limit, thermal shock or pressurized thermal shock to the reactor pressure vessel or overpressurization conditions at low temperatures.

IF CSP-P.1 is entered during a large break loss of coolant accident, it may be exited if RCS pressure is low enough to allow for RHR forward flow. If CSP-P.1 is exited for this reason, then the Potential Loss criteria should not be considered met.

RCS BARRIER THRESHOLDS:

~~1.~~2. RCS or SG Tube Leakage

Loss ~~1~~2.A

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

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

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

Potential Loss ~~1~~2.A

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

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

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

~~Potential Loss 1.B~~

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

~~CSP-P.1 is the Critical Safety Procedure that provides directions to avoid, or limit, thermal shock or pressurized thermal shock to the reactor pressure vessel or overpressurization conditions at low temperatures.~~

~~PWR RCS BARRIER THRESHOLDS:~~

~~Developer Notes:~~

~~Loss 1.A~~

~~Actuation of the ECCS may also be referred to as Safety Injection (SI) actuation or other appropriate site-specific term.~~

~~Potential Loss 1.A~~

~~Depending upon charging pump flow capacities and RCS volume control parameters, developers may use an RCS leak rate value of 50 gpm, or an appropriate site-specific value, as an alternate Potential Loss threshold. If used, the threshold wording should reflect that the determination of the leak rate value excludes normal reductions in RCS inventory (e.g., by the letdown system or RCP seal leakoff).~~

~~Potential Loss 1.B~~

~~Enter the site-specific indications that define an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized). These will typically be parameters and values that would require operators to take prompt action to address a pressurized thermal shock condition. Developers should also determine if the threshold needs to reflect any dependencies used as EOP transition/entry decision points or condition validation criteria (e.g., an EOP used to respond to an excessive RCS cooldown may not be entered or immediately exited if RCS pressure is below a certain value).~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the RCS Integrity Red Path. Because of the complexity of certain decision-points within the Red Path of this CSFST, developers at these plants may elect to not include the specific parameters and values, and instead follow the guidance below.~~

~~Westinghouse ERG Plants~~

~~As a potential loss indication, developers should consider including a threshold the same as, or similar to, "RCS Integrity Red entry conditions met" in accordance with the guidance at the front of this section. As noted above, developers should ensure that the threshold wording reflects any EOP transition/entry decision points or condition validation criteria. For example, a threshold might read "RCS Integrity (P) Red entry conditions met with RCS pressure > 300 psig."~~

~~2. Inadequate Heat Removal~~

~~There is no Loss threshold associated with Inadequate Heat Removal.~~

~~PWR RCS BARRIER THRESHOLDS:~~

~~Potential Loss 2.A~~

~~This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. If CSP-H.1 was entered during a loss of coolant accident, it may be exited if secondary heat sink is not required based on RCS pressure less than non-faulted S/G pressure. If this is the case, then declaration requirements should not be considered to be met.In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.~~

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

~~CSP-H.1 is the Critical Safety Procedure that provides directions if the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a Potential Loss of the RCS barrier.~~

~~**Developer Notes:**~~

~~Potential Loss 2.A~~

~~Enter the site-specific parameters and values that define an extreme challenge to the ability to remove heat from the RCS via the steam generators. These will typically be parameters and values that would require operators to take prompt action to address this condition.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Heat Sink Red Path.~~

~~Westinghouse ERG Plants~~

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

3. RCS Activity / Containment Radiation

Loss 3.A

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

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

RCS BARRIER THRESHOLDS:

~~PWR RCS BARRIER THRESHOLDS:~~

~~Developer Notes:~~

~~Loss 3.A~~

~~The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS activity at Technical Specification allowable limits, into the containment atmosphere. Using RCS activity at~~

~~Technical Specification allowable limits aligns this threshold with IC SU3. Also, RCS activity at this level will typically result in containment radiation levels that can be more readily detected by containment radiation monitors, and more readily differentiated from those caused by piping or component "shine" sources. If desired, a plant may use a lesser value of RCS activity for determining this value.~~

~~In some cases, the site-specific physical location and sensitivity of the containment radiation monitor(s) may be such that radiation from a cloud of released RCS gases cannot be distinguished from radiation emanating from piping and components containing elevated reactor coolant activity. If so, refer to the Developer Notes for Loss/Potential Loss 5.A and determine if an alternate indication is available.~~

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

~~5. Other Indications~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the RCS barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~**RCS BARRIER THRESHOLDS:**~~

~~**Developer Notes:**~~

~~Loss and/or Potential Loss 5.A~~

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

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

~~**PWR RCS BARRIER THRESHOLDS:**~~

~~6.~~5. Emergency Director Judgment

Loss ~~6~~5.A

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

~~**RCS BARRIER THRESHOLDS:**~~

Potential Loss 5~~6~~.A

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

~~**Developer Notes:**~~

~~None~~

~~PWR~~ CONTAINMENT BARRIER THRESHOLDS:

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

1. Critical Safety Function Status Tree

There is no Loss threshold associated with CSFST.

Potential Loss 1.A

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

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

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

~~1.~~2. RCS or SG Tube Leakage

Loss ~~1~~2.A

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

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

CONTAINMENT BARRIER THRESHOLDS:

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU~~3~~4 for the fuel clad barrier (i.e., RCS activity values) and IC SU~~4~~5 for the RCS barrier (i.e., RCS leak rate values).

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

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

~~PWR CONTAINMENT BARRIER THRESHOLDS:~~

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A~~ R ICs.

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

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm ~~(or other value per SU4 Developer Notes)~~	No classification	No classification
Greater than 25 gpm ~~(or other value per SU4 Developer Notes)~~	Unusual Event per SU4	Unusual Event per SU4
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1	Alert per FA1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1	Alert per FA1

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

~~Developer Notes:~~

~~Loss 1.A~~

~~A steam generator power operated relief valve may also be referred to as an atmospheric steam dump valve or other appropriate site-specific term.~~

~~Developers may include an additional site-specific threshold(s) to address prolonged steam releases necessitated by operational considerations if AOPs or EOPs could require that a leaking or RUPTURED steam generator be used to support plant cooldown.~~

~~Developers may wish to consider incorporating the above table into user aids (e.g., a wallboard) or other locations within their basis document.~~

~~PWR~~ CONTAINMENT BARRIER THRESHOLDS:

~~2. Inadequate Heat Removal~~

~~There is no Loss threshold associated with Inadequate Heat Removal.~~

~~Potential Loss 2.A~~

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

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

~~Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.~~

~~Developer Notes:~~

~~Some site-specific EOPs and/or EOP user guidelines may establish decision-making criteria concerning the number or other attributes of thermocouple readings necessary to drive actions (e.g., 5 CETs reading greater than 1,200°F is required before transitioning to an inadequate core cooling procedure). To maintain consistency with EOPs, these decision-making criteria may be used in the core exit thermocouple reading thresholds.~~

~~Potential Loss 2.A.1~~

~~Enter site-specific criteria requiring entry into a core cooling restoration procedure or prompt implementation of core cooling restoration actions. A reading of 1,200°F on the CETs may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Red Path.~~

~~**PWR CONTAINMENT BARRIER THRESHOLDS:**~~

~~Westinghouse ERG Plants~~

~~Developers should consider including a threshold the same as, or similar to, "Core Cooling Red entry conditions met for 15 minutes or longer" in accordance with the guidance at the front of this section.~~

3. RCS Activity / Containment Radiation

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

Potential Loss 3.A

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

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

~~**CONTAINMENT BARRIER THRESHOLDS:**~~

~~**Developer Notes:**~~

~~Potential Loss 3.A~~

~~NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, provides the basis for using the 20% fuel cladding failure value. Unless there is a site-specific analysis justifying a different value, the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad failure into the containment atmosphere.~~

4. Containment Integrity or Bypass

Loss 4.A

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

~~PWR CONTAINMENT BARRIER THRESHOLDS:~~

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

Refer to the middle piping run of Figure 9-F-~~4~~2. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

CONTAINMENT BARRIER THRESHOLDS:

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

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

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

~~CONTAINMENT BARRIER THRESHOLDS:~~

Refer to the top piping run of Figure 9-F-~~4~~2. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

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

~~PWR CONTAINMENT BARRIER THRESHOLDS:~~

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

Refer to the bottom piping run of Figure 9-F-~~4~~2. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump or system piping developed a leak that allowed steam/water to enter the Auxiliary Building, then threshold 4.B would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

CONTAINMENT BARRIER THRESHOLDS:

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

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold ~~1~~2.A.

Loss 4.B

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

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

~~CONTAINMENT BARRIER THRESHOLDS:~~

Refer to the middle piping run of Figure 9-F-2~~4~~. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold 2~~1~~.A to be met.

~~PWR CONTAINMENT BARRIER THRESHOLDS:~~

Potential Loss 4.A

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

CONTAINMENT BARRIER THRESHOLDS:

Potential Loss 44.B

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

Potential Loss 44.C

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

During a design basis accident, a minimum of two CONTAINMENT Accident Fan Cooler Units with their accident fans running and one CONTAINMENT spray train are required to maintain the CONTAINMENT peak pressure and temperature below the design limits. Each CONTAINMENT Spray train is a CONTAINMENT spray pump, spray header, nozzles, valves and piping. Each CONTAINMENT Accident Fan Cooler Unit consists of cooling coils, accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path.

~~**Developer Notes:**~~

~~Loss 4.A.1~~

~~Developers may include a list of site-specific radiation monitors to better define this threshold. Expected monitor alarms or readings may also be included.~~

~~Potential Loss 4.A~~

~~The site-specific pressure is the containment design pressure.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, the pressure value in Potential Loss 4.A is that used for the Containment Red Path. If the Containment CSFST contains more than one Red Path due to other dependencies (e.g., status of containment isolation), enter the highest containment pressure value shown on the tree. This is typically the containment design pressure.~~

~~**PWR CONTAINMENT BARRIER THRESHOLDS:**~~

~~Potential Loss 4.B~~

~~Developers may enter the minimum containment atmospheric hydrogen concentration necessary to support a hydrogen burn (i.e., the lower deflagration limit). A concurrent containment oxygen concentration may be included if the plant has this indication available in the Control Room.~~

~~Potential Loss 4.C~~

~~Enter the site-specific pressure setpoint value that actuates containment pressure control systems (e.g., containment spray). Also enter the site-specific containment pressure control system/equipment that should be operating per design if the containment pressure setpoint is reached. If desired, specific condition indications such as parameter values can also be entered (e.g., a containment spray flow rate less than a certain value).~~

~~This threshold is not applicable to the U.S. Evolutionary Power Reactor (EPR) design.~~

~~Westinghouse ERG Plants~~

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

~~5. Other Indications~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Containment barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~CONTAINMENT BARRIER THRESHOLDS:~~

~~Developer Notes:~~

~~Loss and/or Potential Loss 5.A~~

~~If site emergency operating procedures provide for venting of the containment as a means of preventing catastrophic failure, a Loss threshold should be included for the containment barrier. This threshold would be met as soon as such venting is IMMINENT. Containment venting as part of recovery actions is classified in accordance with the radiological effluent ICs.~~

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

~~PWR CONTAINMENT BARRIER THRESHOLDS:~~

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

~~6.~~5. Emergency Director Judgment

Loss ~~6~~5.A

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

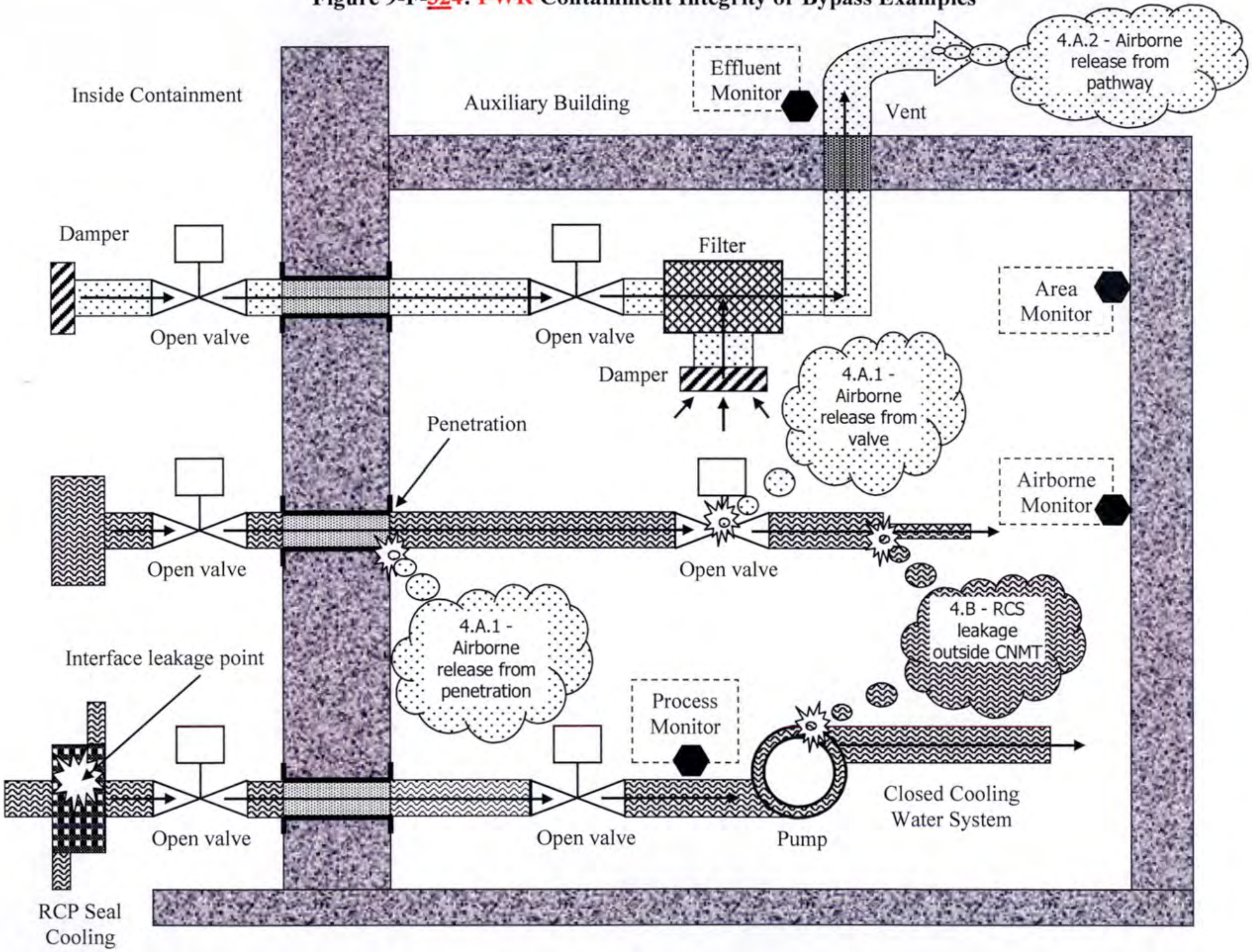
Potential Loss ~~6~~5.A

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

~~**Developer Notes:**~~

~~None~~

Figure 9-F-~~3~~24: ~~PWR~~ Containment Integrity or Bypass Examples



10 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

~~Table H-1: Recognition Category "H" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~**HU1** Confirmed SECURITY CONDITION or threat.~~ ~~*Op. Modes: All*~~	~~**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.~~ ~~*Op. Modes: All*~~	~~**HS1** HOSTILE ACTION within the PROTECTED AREA.~~ ~~*Op. Modes: All*~~	~~**HG1** HOSTILE ACTION resulting in loss of physical control of the facility.~~ ~~*Op. Modes: All*~~
~~**HU2** Seismic event greater than OBE levels.~~ ~~*Op. Modes: All*~~			
~~**HU3** Hazardous event.~~ ~~*Op. Modes: All*~~			
~~**HU4** FIRE potentially degrading the level of safety of the plant.~~ ~~*Op. Modes: All*~~			
	~~**HA5** Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.~~ ~~*Op. Modes: All*~~		
	~~**HA6** Control Room evacuation resulting in transfer of plant control to alternate locations.~~ ~~*Op. Modes: All*~~	~~**HS6** Inability to control a key safety function from outside the Control Room.~~ ~~*Op. Modes: All*~~	
~~**HU7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE.~~ ~~*Op. Modes: All*~~	~~**HA7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.~~ ~~*Op. Modes: All*~~	~~**HS7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.~~ ~~*Op. Modes: All*~~	~~**HG7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.~~ ~~*Op. Modes: All*~~

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

HU1

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: Confirmed SECURITY CONDITION or threat.

Operating Mode Applicability: All

Emergency Action Levels:

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

HU1.1 A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the ~~(site-specific s~~Security ~~shift~~ Shift ~~supervision~~Supervisor~~)~~.

HU1.2 Notification of a credible security threat directed at ~~the site~~PBNP.

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

Definitions:

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

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

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

Basis:

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

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

these events will initiate appropriate threat-related notifications to plant personnel and ~~ORO~~offsite response organizations.

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

EAL HU1.1 references ~~(site-specific s~~Security ~~shift~~ Shift ~~supervision~~Supervisor~~)~~ because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.390 information.

EAL HU1.2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with SY-AA-102-1014, Threat Assessment and Reporting~~(site-specific procedure)~~.

EAL HU1.3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with ~~(site-specific procedure)~~AOP-29, Security Threat.

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

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

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~The (site-specific procedure) is the procedure(s) used by Control Room and/or Security personnel to determine if a security threat is credible, and to validate receipt of aircraft threat information.~~

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

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~ECL Assignment Attributes: 3.1.1.A~~

HU2

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: Seismic event greater than OBE levels.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HU2.1 Seismic event greater than Operating Basis Earthquake (OBE) as indicated by seismic monitor indication of ground acceleration greater than:

- -0.06 g horizontal

OR

- 0.04 g vertical.

~~(1) :~~

~~(site-specific indication that a seismic event met or exceeded OBE limits)~~

Definitions:

None

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE)[1]. An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE)[2] should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event ~~(e.g., typical lateral accelerations are in excess of 0.08g)~~. The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

[1] An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.

[2] An SSE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

~~**Developer Notes:**~~

~~This "site-specific indication that a seismic event met or exceeded OBE limits" should be based on the indications, alarms and displays of site-specific seismic monitoring equipment.~~

~~Indications described in the EAL should be limited to those that are immediately available to Control Room personnel and which can be readily assessed. Indications available outside the Control Room and/or which require lengthy times to assess (e.g., processing of scratch plates or recorded data) should not be used. The goal is to specify indications that can be assessed within 15-minutes of the actual or suspected seismic event.~~

~~For sites that do not have readily assessable OBE indications within the Control Room, developers should use the following alternate EAL (or similar wording).~~

~~(1) a. Control Room personnel feel an actual or potential seismic event.~~

~~**AND**~~

~~b. The occurrence of a seismic event is confirmed in manner deemed appropriate by the Shift Manager or Emergency Director.~~

~~The EAL 1.b statement is included to ensure that a declaration does not result from felt vibrations caused by a non-seismic source (e.g., a dropped heavy load). The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration. It is recognized that this alternate EAL wording may cause a site to declare an Unusual Event while another site, similarly affected but with readily assessable OBE indications in the Control Room, may not.~~

~~The above alternate wording may also be used to develop a compensatory EAL for use during periods when a seismic monitoring system capable of detecting an OBE is out-of-service for maintenance or repair.~~

~~ECL Assignment Attributes: 3.1.1.A~~

HU3

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: Hazardous events

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2 or 3 or 4 or 5 or 6)~~

Note: EAL HU3.~~3~~4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

HU3.1 A tornado strike within the PROTECTED AREA.

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

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

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

HU3.5 ~~(Site-specific list of natural or technological hazard events)~~Lake level greater than or equal to +8.0 ft. (Plant elevation).

HU3.6 Pump bay level less than -15.0 ft.

Definitions:

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

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

Basis:

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

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

EAL HU3.2 addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation

of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

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

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

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

EAL HU3.5 addresses lake water level as the Turbine Building is susceptible to external flooding. +8.0 ~~ft.corresponds~~ft. corresponds to the Turbine Building floor elevation~~(site-specific description)~~.

EAL HU3.6 addresses operability of the Emergency Diesel Generators and all Containment fan coolers. The pump bay level of -15.0 ~~ft.represents~~ft. represents the value at which the Emergency Diesel Generators and all Containment Fan Coolers must be declared inoperable, and is four feet above the level at which the service water pumps may begin to cavitate.

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

~~**Developer Notes:**~~

~~The "Site-specific list of natural or technological hazard events" should include other events that may be a precursor to a more significant event or condition, and that are appropriate to the site location and characteristics.~~

~~Notwithstanding the events specifically included as EALs above, a "Site-specific list of natural or technological hazard events" need not include short-lived events for which the extent of the damage and the resulting consequences can be determined within a relatively short time frame. In these cases, a damage assessment can be performed soon after the event, and the plant staff will be able to identify potential or actual impacts to plant systems and structures. This will enable prompt definition and implementation of compensatory or corrective measures with no appreciable increase in risk to the public.~~

~~To the extent that a short-lived event does cause immediate and significant damage to plant systems and structures, it will be classifiable under the Recognition Category F, S and C ICs and EALs. Events of lesser impact would be expected to cause only small and localized damage. The consequences from these types of events are adequately assessed and addressed in accordance with Technical Specifications. In addition, the occurrence or effects of the event may be reportable under the requirements of 10 CFR 50.72.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1.C~~

HU4

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2 or 3 or 4)~~

Notes:

- The Emergency Director 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) on the FACP system (this note is applicable in Modes 1 and 2 only).

HU4.1 a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications:

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

b. The FIRE is located within **ANY** ~~of the following~~Table H-1 plant rooms or areas~~:~~

~~(site-specific list of plant rooms or areas)~~

HU4.2 a. Receipt of a single fire alarm ~~(i.e.,~~with no other indications of a FIRE~~)~~.

AND

b. The FIRE is located within **ANY** ~~of the following~~Table H-1 plant rooms or areas except Containment in Modes 1 and 2 (see Note above):

~~(site-specific list of plant rooms or areas)~~

AND

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

HU4.3 A FIRE within the plant or ISFSI ~~*[for plants with an ISFSI outside the plant Protected Area]*~~ PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

HU4.4 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.

Table H-1 Areas
Control Room
Containment
PAB

G05 building
13.8kV Building
Cable Spreading Room
Vital Switchgear Room
AFW Pump Room
G-01/02 Rooms
EDG Building
Service Water Pump Rooms
Façade 85'

Definitions:

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

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

Basis:

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

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

EAL HU4.1

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

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

EAL HU4.2

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

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. Except for the Containment Building in Modes 1 or 2, t~~T~~he 30-~~-~~minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

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

EAL HU4.3

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

EAL HU4.4

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

Basis-Related Requirements from Appendix R and NFPA-805

Criterion 3 of Appendix A to 10 CFR 50 states in part that "structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

The Nuclear Safety Goal ("NSG") in NFPA 805, Section 1.3.1 states, "The nuclear safety goal is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance because a safe shutdown success path, free of fire damage, must be available to meet the nuclear safety goals, objectives and performance criteria for a fire under any plant operational mode or configuration.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

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

~~Appendix R to 10 CFR 50, states in part:~~

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

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

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

~~**Developer Notes:**~~

~~The "site-specific list of plant rooms or areas" should specify those rooms or areas that contain SAFETY SYSTEM equipment.~~

~~As noted in the EALs and Basis section, include the term ISFSI if the site has an ISFSI outside the plant Protected Area.~~

~~ECL Assignment Attributes: 3.1.1.A~~

HU7

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a ~~(NO)~~UE.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HU7.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS ~~safety systems~~ occurs.

Definitions:

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

Basis:

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

HA1

ECL: Alert

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

Operating Mode Applicability: All

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

HA1.1 A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the ~~(site-specific s~~Security ~~shift~~ Shift ~~supervision~~Supervisor~~)~~.

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

Definitions:

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

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

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

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

Basis:

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

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

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

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

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

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

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

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

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

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

EAL HA1.2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and ~~ORO~~offsite response organizations are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with ~~(~~SY-AA-102-1014, Threat Assessment and Reporting~~site-specific procedure).~~s.

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

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

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

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

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

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

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the OWNER CONTROLLED AREA.~~

~~ECL Assignment Attributes: 3.1.2.D~~

HA5

ECL: Alert

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

Operating Mode Applicability: All

~~Example~~ 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.

HA5.1 a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas:

Area	Mode
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B32 MCC Area	4

~~(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.

Definitions:

None

Basis:

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

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

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

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

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

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

The list of plant rooms or areas in EAL HA5.1 was generated from a step-by-step review of OP-3A, 3B, 3C, 5D, and 7A.

~~This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area, or to intentional inerting of containment (BWR only).~~

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

~~This list was generated from a step-by-step review of OP-3A, B, and C.~~

~~**Developer Notes:**~~

~~The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.~~

~~The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~

~~The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.~~

~~If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.~~

~~ECL Assignment Attributes: 3.1.2.B~~

HA6

ECL: Alert

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HA6.1 An event has resulted in plant control being transferred from the Control Room to ~~(site-specific remote shutdown panels and local control stations)~~AOP local control stations.

Definitions:

None

Basis:

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

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

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

~~**Developer Notes:**~~

~~The "site-specific remote shutdown panels and local control stations" are the panels and control stations referenced in plant procedures used to cooldown and shutdown the plant from a location(s) outside the Control Room.~~

~~ECL Assignment Attributes: 3.1.2.B~~

HA7

ECL: Alert

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

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Levels:

HA7.1 Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Definitions:

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

Basis:

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

HS1

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Levels:

~~H~~HS1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the ~~(site-specific s~~Security ~~shift~~ Shift ~~supervision~~Supervisor~~)~~.

Definitions:

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

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

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

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

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

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

Basis:

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

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

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

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

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

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

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

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

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

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the PROTECTED AREA.~~

~~ECL Assignment Attributes: 3.1.3.D~~

HS6

ECL: Site Area Emergency

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that ~~(site-specific number of~~15 minutes~~)~~ has been exceeded, or will likely be exceeded.

HS6.1 a. An event has resulted in plant control being transferred from the Control Room to ~~(site-specific remote shutdown panels and~~AOP local control stations~~)~~.

AND

b. Control of **ANY** of the following key safety functions is not reestablished within ~~(site-specific number of~~15 minutes~~)~~.

- Reactivity control
- Core cooling ~~[*PWR*] / RPV water level [*BWR*]~~
- RCS heat removal

Definitions:

None

Basis:

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

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within ~~(the site-specific time for transfer)~~ 15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

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

~~Developer Notes:~~

~~The "site-specific remote shutdown panels and local control stations" are the panels and control stations referenced in plant procedures used to cooldown and shutdown the plant from a location(s) outside the Control Room.~~

~~The "site-specific number of minutes" is the time in which plant control must be (or is expected to be) reestablished at an alternate location as described in the site-specific fire response analyses. Absent a basis in the site-specific analyses, 15 minutes should be used. Another time period may be used with appropriate basis/justification.~~

~~ECL Assignment Attributes: 3.1.3.B~~

HS7

ECL: Site Area Emergency

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HS7.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Definitions:

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

Basis:

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

HG1

ECL: General Emergency

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HG1.1 a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the ~~(site-specific s~~Security ~~shift~~ Shift ~~supervision~~Supervisor~~)~~.

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.

Definitions:

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

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

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

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

Basis:

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

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

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

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

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

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

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the PROTECTED AREA.~~

~~ECL Assignment Attributes: 3.1.4.D~~

HG7

ECL: General Emergency

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HG7.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Definitions:

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

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

Basis:

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

11 SYSTEM MALFUNCTION ICS/EALS

~~**Table S-1: Recognition Category "S" Initiating Condition Matrix**~~

~~**UNUSUAL EVENT**~~	~~**ALERT**~~	~~**SITE AREA EMERGENCY**~~	~~**GENERAL EMERGENCY**~~
~~**SU1** Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	~~**SA1** Loss of all but one AC power source to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	~~**SS1** Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	~~**SG1** Prolonged loss of all offsite and all onsite AC power to emergency buses.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~
~~**SU2** UNPLANNED loss of Control Room indications for 15 minutes or longer.~~ ~~*Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown1, 2, 3, 4*~~	~~**SA2** UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.~~ ~~*Op. Modes: 1, 2, 3, 4 Power Operation, Startup, Hot Standby, Hot Shutdown*~~		
~~**SU3** Reactor coolant activity greater than Technical Specification allowable limits.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~			
~~**SU4** RCS leakage for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~			
~~**SU5** Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor.~~ ~~*Op. Modes: Power Operation1*~~	~~**SA5** Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.~~ ~~*Op. Modes: Power Operation1*~~	~~**SS5** Inability to shutdown the reactor causing a challenge to (core cooling [*PWR*] / RPV water level [*BWR*]) or RCS heat removal.~~ ~~*Op. Modes: Power Operation1*~~	

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~**SU6** Loss of all onsite or offsite communications capabilities.~~ *~~Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown~~*			
~~**SU7** Failure to isolate containment or loss of containment pressure control. [*PWR*]~~ *~~Op. Modes: 1, 2, 3, 4 Power Operation, Startup, Hot Standby, Hot Shutdown~~*			
		~~**SS8** Loss of all Vital DC power for 15 minutes or longer.~~ *~~Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown~~*	~~**SG8** Loss of all AC and Vital DC power sources for 15 minutes or longer.~~ *~~Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown~~*
	~~**SA9** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.~~ *~~Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown~~*		

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

SU1

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~**Example**~~ **Emergency Action Levels:**

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

SU1.1 Loss of **ALL** offsite AC power capability to ~~(site-specific emergency buses)~~1(2)-A-05 and 1(2)-A-06 for 15 minutes or longer.

Definitions:

None

Basis:

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

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

Note: with respect to this EAL, "Station Blackout is Unit 1(2) specific."

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

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

~~**Developer Notes:**~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an~~

~~affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~SU2~~SU3

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~Example~~ Emergency Action Levels:

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

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

- Reactor Power
- RCS / Pressurizer Level
- RCS / Pressurizer Pressure
- Core Exit / RCS Temperature
- Level in at least one steam generator
- Steam Generator Auxiliary Feed Water Flow

~~[*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~~

Definitions:

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

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

Basis:

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

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

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

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

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

Escalation of the emergency classification level would be via IC SA~~2~~3.

~~**Developer Notes:**~~

~~In the PWR parameter list column, the "site-specific number" should reflect the minimum number of steam generators necessary for plant cooldown and shutdown. This criterion may also specify whether the level value should be wide-range, narrow-range or both, depending upon the monitoring requirements in emergency operating procedures.~~

~~Developers may specify either pressurizer or reactor vessel level in the PWR parameter column entry for RCS Level.~~

~~The number, type, location and layout of Control Room indications, and the range of possible failure modes, can challenge the ability of an operator to accurately determine, within the time period available for emergency classification assessments, if a specific percentage of indications have been lost. The approach used in this EAL facilitates prompt and accurate emergency classification assessments by focusing on the indications for a selected subset of parameters.~~

~~By focusing on the availability of the specified parameter values, instead of the sources of those values, the EAL recognizes and accommodates the wide variety of indications in nuclear power plant Control Rooms. Indication types and sources may be analog or digital, safety-related or not, primary or alternate, individual meter value or computer group display, etc.~~

~~A loss of plant annunciators will be evaluated for reportability in accordance with 10 CFR 50.72 (and the associated guidance in NUREG-1022), and reported if it significantly impairs the capability to perform emergency assessments. Compensatory measures for a loss of annunciation can be readily implemented and may include increased monitoring of main control boards and more frequent plant rounds by non-licensed operators. Their alerting function notwithstanding, annunciators do not provide the parameter values or specific component status information used to operate the plant, or process through AOPs or EOPs. Based on these considerations, a loss of~~

~~annunciation is considered to be adequately addressed by reportability criteria, and therefore not included in this IC and EAL.~~

~~With respect to establishing event severity, the response to a loss of radiation monitoring data (e.g., process or effluent monitor values) is considered to be adequately bounded by the requirements of 10 CFR 50.72 (and associated guidance in NUREG-1022). The reporting of this event will ensure adequate plant staff and NRC awareness, and drive the establishment of appropriate compensatory measures and corrective actions. In addition, a loss of radiation monitoring data, by itself, is not a precursor to a more significant event.~~

~~Personnel at sites that have a Failure Modes and Effects Analysis (FMEA) included within the design basis of a digital I&C system should consider the FMEA information when developing their site-specific EALs.~~

~~Due to changes in the configurations of SAFETY SYSTEMS, including associated instrumentation and indications, during the cold shutdown, refueling, and defueled modes, no analogous IC is included for these modes of operation.~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~SU3~~SU4

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~**Example**~~ **Emergency Action Levels:** ~~(1 or 2)~~

SU4.1 ~~(Site-specific radiation monitor)~~Failed Fuel Monitor 1(2)-RE-109 reading greater than ~~(site-specific value)~~750 mR/hr.

SU4.2 Sample analysis indicates that a RCS Specific Activity value is greater than an allowable limit specified in Technical Specifications as indicated by **ANY** of the following conditions:

a. Dose Equivalent I-131 greater than 50 μCi/gm

OR

b. Dose Equivalent I-131 greater than 0.5 μCi/gm but less than or equal to 50 μCi/gm for greater than 48 hours

OR

c. Dose Equivalent Xe-133 greater than 300 μCi/gm for greater than 48 hours~~Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.~~

Definitions:

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specification~~s~~ 3.4.16 for longer than the allowed completion time. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

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

~~**Developer Notes:**~~

~~For EAL #1 – Enter the radiation monitor(s) that may be used to readily identify when RCS activity levels exceed Technical Specification allowable limits. This EAL may be developed using different methods and sites should use existing capabilities to address it (e.g., development of new capabilities is not required). Examples of existing methods/capabilities include:~~

- ~~An installed radiation monitor on the letdown system or air ejector.~~
- ~~A hand-held monitor or deployed detector reading with pre-calculated conversion values or readily implementable conversion calculation capability.~~

~~The monitor reading values should correspond to an RCS activity level approximately at Technical Specification allowable limits.~~

~~If there is no existing method/capability for determining this EAL, then it should not be included. IC evaluation will be based on EAL #2.~~

~~For EAL#2 – Developers may reword the EAL to include the reactor coolant activity parameter(s) specified in Technical Specifications and the associated allowable limit(s) (e.g., values for dose equivalent I-131 and gross activity, time-dependent or transient values, etc.). If this approach is selected, all RCS activity allowable limits should be included.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1.B~~

~~SU4~~SU5

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

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

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

SU5.1 RCS unidentified or pressure boundary leakage greater than ~~(site-specific value)~~10 gpm for 15 minutes or longer.

SU5.2 RCS identified leakage greater than ~~(site-specific value)~~25 gpm for 15 minutes or longer.

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

Definitions:

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

Basis:

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

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

EAL SU5.3 addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These EALs thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage ~~in a PWR~~) or a location outside of containment.

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

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. For ~~PWRs~~PBNP, an emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).~~ For BWRs, a stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.~~

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

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

~~**Developer Notes:**~~

~~EAL #1 – For the site-specific leak rate value, enter the higher of 10 gpm or the value specified in the site's Technical Specifications for this type of leakage.~~

~~EAL #2 – For the site-specific leak rate value, enter the higher of 25 gpm or the value specified in the site's Technical Specifications for this type of leakage.~~

~~For sites that have Technical Specifications that do not specify a leakage type for steam generator tube leakage, developers should include an EAL for tube leakage greater than 25 gpm for 15 minutes or longer.~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~SU5~~SU6

ECL: ~~Notification of Unusual Event~~Unusual Event

Initiating Condition: Automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ fails to shutdown the reactor.

Operating Mode Applicability: ~~Power Operation~~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.~~

~~**Example**~~ **Emergency Action Levels:** ~~(1 or 2)~~

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

SU6.1

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.

SU6.2 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.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~ that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~ 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.

NOTE: For PBNP, the phrase "at the reactor control consoles" means the reactor trip pushbuttons on the following panels:

- Unit 1 on panels 1C04 and C01
- Unit 2 on panels 2C04 and C02

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

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

~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [BWR]~~

The plant response to the failure of an automatic or manual reactor ~~(~~trip~~ [PWR] / scram [BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA~~5~~6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA~~5~~6 or FA1, an Unusual Event declaration is appropriate for this event.

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

Should a reactor ~~(~~trip~~ [PWR] / scram [BWR])~~ signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

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

~~Developer Notes:~~

~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode.~~

~~Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).~~

~~The term "reactor control consoles" may be replaced with the appropriate site-specific term (e.g., main control boards).~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~SU6~~SU7

ECL: ~~Notification of Unusual Event~~Unusual Event

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

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

SU7.1 Loss of **ALL** of the following onsite communication methods:
- Plant Public Address System (Gai-Tronics)
- Commercial Phones
- PBX Phones
- Security Radio
- Portable Radios

SU7.2 Loss of **ALL** of the following offsite response organization communications methods:
- Nuclear Accident Reporting System (NARS)
- Commercial Phones
- PBX Phones
- Satellite Phones
- Manitowoc County Sheriff's Department Radio

SU7.3 Loss of **ALL** of the following NRC communications methods:
- FTS Phone System
- Commercial Phones
- PBX Phones
- Satellite Phones

Definitions:

None

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 ~~ORO~~offsite response organizations and the NRC.

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

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

EAL SU7.2 addresses a total loss of the communications methods used to notify all ~~ORO~~offsite response organizations of an emergency declaration. The ~~ORO~~offsite response

organizations referred to here are ~~(see Developer Notes)~~the State of Wisconsin, Manitowoc County, and Kewaunee County.

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

~~**Developer Notes:**~~

~~EAL #1 - The "site-specific list of communications methods" should include all communications methods used for routine plant communications (e.g., commercial or site telephones, page-party systems, radios, etc.). This listing should include installed plant equipment and components, and not items owned and maintained by individuals.~~

~~EAL #2 - The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to OROs as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. Example methods are ring-down/dedicated telephone lines, commercial telephone lines, radios, satellite telephones and internet-based communications technology.~~

~~In the Basis section, insert the site-specific listing of the OROs requiring notification of an emergency declaration from the Control Room in accordance with the site Emergency Plan, and typically within 15 minutes.~~

~~EAL #3 - The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to the NRC as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. These methods are typically the dedicated Emergency Notification System (ENS) telephone line and commercial telephone lines.~~

~~ECL Assignment Attributes: 3.1.1.C~~

~~SU7~~SU8

ECL: ~~Notification of~~ Unusual Event

Initiating Condition: Failure to isolate containment or loss of containment pressure control. ~~[*PWR*]~~

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

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

SU8.1 a. Failure of containment to isolate when required by an actuation signal.

AND

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

SU8.2 a. Containment pressure greater than ~~(site-specific pressure)~~25 psig.

AND

b. Less than one full train of ~~(site-specific system or equipment)~~Containment Cooling System equipment is operating per design for 15 minutes or longer.

Definitions:

None

Basis:

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

For EAL SU8.1, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

EAL SU8.2 addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. During a design basis accident, a minimum of two containment accident fan cooler units with their accident fans running and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits. Each containment spray train is a containment spray pump, spray header, nozzles, valves and piping. Each containment accident fan cooler unit consists of cooling coils,

accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path.

The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems ~~(e.g., containment sprays or ice condenser fans)~~ are either lost or performing in a degraded manner.

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

~~**Developer Notes:**~~

~~Enter the "site-specific pressure" value that actuates containment pressure control systems (e.g., containment spray). Also enter the site-specific containment pressure control system/equipment that should be operating per design if the containment pressure actuation setpoint is reached. If desired, specific condition indications such as parameter values can also be entered (e.g., a containment spray flow rate less than a certain value).~~

~~EAL #2 is not applicable to the U.S. Evolutionary Power Reactor (EPR) design.~~

~~ECL Assignment Attributes: 3.1.1.A~~

SA1

ECL: Alert

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~Example Emergency~~Emergency Action Levels:

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

SA1.1 a. AC power capability to ~~(site-specific emergency buses)~~1(2)-A-05 **AND** 1(2)-A-06 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~~ **ALL** AC power to SAFETY SYSTEMS.

Definitions:

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

Basis:

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

- Normal Unit 1(2) offsite power sources include:
  - 345 KVAC 1(2)X-03 through the 13.8 KVAC system to the LVSAT 1(2)X-04
  - 345 KVAC backfed through the 19 KVAC system to the UAT 1(2)X-02

- Normal Unit 1(2) onsite power sources consist of:
  - emergency diesel generators
  - gas turbine generator
  - unit main turbine generator
  - power supplied from the opposite unit

An "AC power source" is a source recognized in AOPs and EOPs (including Beyond Design Basis event procedures), ~~,~~ and capable of supplying required power to an emergency bus. Some examples of this Initiating C~~c~~ondition are presented below.

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

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

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

~~**Developer Notes:**~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide required power to an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50%-capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~Developers should modify the bulleted examples provided in the basis section, above, as needed to reflect their site-specific plant designs and capabilities.~~

~~The EALs and Basis should reflect that each independent offsite power circuit constitutes a single power source. For example, three independent 345kV offsite power circuits (i.e., incoming power lines) comprise three separate power sources. Independence may be determined from a review of the site-specific UFSAR, SBO analysis or related loss of electrical power studies.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is recognized in AOPs and EOPs, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.2.B~~

~~SA2~~SA3

ECL: Alert

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~Example~~ Emergency Action Levels:

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

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

- Reactor Power
- RCS / Pressurizer Level
- RCS / Pressurizer Pressure
- Core Exit / RCS Temperature
- Levels in at least one steam generator
- Steam Generator Auxiliary Feed Water Flow

~~*[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*]~~

Definitions:

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

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

Basis:

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

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

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

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

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

Escalation of the emergency classification level would be via ICs FS1 or IC ~~AS1~~RS1.

~~**Developer Notes:**~~

~~In the PWR parameter list column, the "site-specific number" should reflect the minimum number of steam generators necessary for plant cooldown and shutdown. This criterion may also specify whether the level value should be wide-range, narrow-range or both, depending upon the monitoring requirements in emergency operating procedures.~~

~~Developers may specify either pressurizer or reactor vessel level in the PWR parameter column entry for RCS Level.~~

~~Developers should consider if the "transient events" list needs to be modified to better reflect site-specific plant operating characteristics and expected responses.~~

~~The number, type, location and layout of Control Room indications, and the range of possible failure modes, can challenge the ability of an operator to accurately determine, within the time period available for emergency classification assessments, if a specific percentage of indications have been lost. The approach used in this EAL facilitates prompt and accurate emergency classification assessments by focusing on the indications for a selected subset of parameters.~~

~~By focusing on the availability of the specified parameter values, instead of the sources of those values, the EAL recognizes and accommodates the wide variety of indications in nuclear power plant Control Rooms. Indication types and sources may be analog or digital, safety-related or not, primary or alternate, individual meter value or computer group display, etc.~~

~~A loss of plant annunciators will be evaluated for reportability in accordance with 10 CFR 50.72 (and the associated guidance in NUREG-1022), and reported if it significantly impairs the capability to perform emergency assessments. Compensatory measures for a loss of annunciation can be readily implemented and may include increased monitoring of main control boards and more frequent plant rounds by non-licensed operators. Their alerting function notwithstanding, annunciators do not provide the parameter values or specific component status information used to operate the plant, or process through AOPs or EOPs. Based on these considerations, a loss of annunciation is considered to be adequately addressed by reportability criteria, and therefore not included in this IC and EAL.~~

~~With respect to establishing event severity, the response to a loss of radiation monitoring data (e.g., process or effluent monitor values) is considered to be adequately bounded by the requirements of 10 CFR 50.72 (and associated guidance in NUREG-1022). The reporting of this event will ensure adequate plant staff and NRC awareness, and drive the establishment of appropriate compensatory measures and corrective actions. In addition, a loss of radiation monitoring data, by itself, is not a precursor to a more significant event.~~

~~Personnel at sites that have a Failure Modes and Effects Analysis (FMEA) included within the design basis of a digital I&C system should consider the FMEA information when developing their site-specific EALs.~~

~~Due to changes in the configurations of SAFETY SYSTEMS, including associated instrumentation and indications, during the cold shutdown, refueling, and defueled modes, no analogous IC is included for these modes of operation.~~

~~ECL Assignment Attributes: 3.1.2.B~~

~~SA5~~SA6

ECL: Alert

Initiating Condition: Automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.

Operating Mode Applicability: ~~Power Operation~~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.

~~Example Emergency~~Emergency Action Levels:

S~~A~~6.1 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.

Definitions:

None

Basis:

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

NOTE: For PBNP, the phrase "at the reactor control consoles" means the reactor trip pushbuttons on the following panels:

- Unit 1 on panels 1C04 and C01
- Unit 2 on panels 2C04 and C02

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

~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [*BWR*]~~

The plant response to the failure of an automatic or manual reactor ~~(trip [*PWR*] / scram [*BWR*])~~ 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 ~~[PWR] / RPV water level [BWR]~~ or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS~~5~~6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS~~5~~6 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.

~~**Developer Notes:**~~

~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode.~~

~~Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).~~

~~The term "reactor control consoles" may be replaced with the appropriate site-specific term (e.g., main control boards).~~

~~ECL Assignment Attributes: 3.1.2.B~~

SA9

ECL: Alert

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~**Example**~~ **Emergency Action Levels:**

Notes:

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

SA9.1 a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- ~~(site-specific hazards)~~Lake level greater than or equal to +9.0 ft. (Plant elevation)
- Pump bay level less than -19.0 ft.
- Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

AND

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

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Definitions:

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

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

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

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

~~. **EITHER** of the following:~~

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

~~**OR**~~

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

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria SA9.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded

performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

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

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

High lake water level conditions that may have resulted in a plant VITAL AREA being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant SAFETY SYSTEMS. Lake water level at +9.0 feet corresponds to the license basis flood elevation and is one foot above the Turbine Building floor elevation. The low pump bay level ~~setpoint~~threshold corresponds to the level that is calculated to correspond to the onset of cavitation of the service water pumps.

~~EAL 1.b.1 addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.~~

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

Escalation of the emergency classification level would be via IC FS1 or ~~AS1~~RS1.

~~**Developer Notes:**~~

~~For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).~~

~~Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.~~

~~ECL Assignment Attributes: 3.1.2.B~~

SS1

ECL: Site Area Emergency

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~**Example**~~ **Emergency Action Levels:**

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

S~~S~~1.1 Loss of **ALL** offsite and **ALL** onsite AC power to ~~(site-specific emergency buses)~~1(2)-A-05 and 1(2)-A-06 for 15 minutes or longer.

Definitions:

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

Basis:

For the purpose of classification under this EAL, evaluation of power sources should be made on each unit individually.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to safety-related 4160 VAC busses. Even though a safety-related 4160 VAC bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.

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

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or SG1.

~~**Developer Notes:**~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide adequate power to an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50%-capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.3.B~~

SS2

ECL: Site Area Emergency

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

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

SS2.1 Indicated voltage is less than 115 VDC on **ALL** Vital DC busses D-01, D-02, D-03, and D-04 for 15 minutes or longer.

Definitions:

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

Basis:

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

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

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

~~SS5~~SS6

ECL: Site Area Emergency

Initiating Condition: Inability to shutdown the reactor causing a challenge to ~~(~~core cooling ~~[*PWR*] / RPV water level [*BWR*])~~ or RCS heat removal.

Operating Mode Applicability: ~~Power Operation~~1

~~Example~~ Emergency Action Levels:

~~SS~~S6.1 a. An automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ 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:

- ~~(Site-specific indication of an inability to adequately remove heat from the core)~~Conditions requiring entry into Core Cooling – Red Path (CSP-C.1) are met.
- ~~(Site-specific indication of an inability to adequately remove heat from the RCS)~~Conditions requiring entry into Heat Sink – Red Path (CSP-H.1) are met.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~ 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 ~~AG1~~ RG1 or FG1.

~~**Developer Notes:**~~

~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode.~~

~~Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).~~

~~Site-specific indication of an inability to adequately remove heat from the core:~~

~~[*BWR*] – Reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (as described in the EOP bases).~~

~~[*PWR*] – Insert site-specific values for an incore/core exit thermocouple temperature and/or reactor vessel water level that drives entry into a core cooling restoration procedure (or otherwise requires implementation of prompt restoration actions). Alternately, a site may use incore/core exit thermocouple temperatures greater than 1,200°F and/or a reactor vessel water level that corresponds to approximately the middle of active fuel. Plants with reactor vessel level instrumentation that cannot measure down to approximately the middle of active fuel should use the lowest on-scale reading that is not above the top of active fuel. If the lowest on-scale reading is above the top of active fuel, then a reactor vessel level value should not be included.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters used in the Core Cooling Red Path.~~

~~Site-specific indication of an inability to adequately remove heat from the RCS:~~

~~[*BWR*] - Use the Heat Capacity Temperature Limit. This addresses the inability to remove heat via the main condenser and the suppression pool due to high pool water temperature.~~

~~[*PWR*] - Insert site-specific parameters associated with inadequate RCS heat removal via the steam generators. These parameters should be identical to those used for the Inadequate Heat Removal threshold Fuel Clad Barrier Potential Loss 2.B and threshold RCS Barrier Potential Loss 2.A in the PWR EAL Fission Product Barrier Table.~~

~~ECL Assignment Attributes: 3.1.3.B~~

~~SS8~~

~~**ECL:** Site Area Emergency~~

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

~~**Operating Mode Applicability:** Power Operation, Startup, Hot Standby, Hot Shutdown1, 2, 3, 4~~

~~**Example Emergency Action Levels:**~~

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

~~1 Indicated voltage is less than (site-specific bus voltage value)115 VDC on **ALL** (site-specific Vital DC busses)1(2)-D-01, D-02, D-03, and D-04 for 15 minutes or longer.~~

~~**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. These systems are classified as safety-related.~~

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

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

~~Escalation of the emergency classification level would be via ICs AG1RG1, FG1 or SG8.~~

~~**Developer Notes:**~~

~~The "site-specific bus voltage value" should be based on the minimum bus voltage necessary for adequate operation of SAFETY SYSTEM equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed.~~

~~The typical value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries, the cell voltage is approximately 1.75 Volts per cell. For a 58 string battery set, the minimum voltage is approximately 1.81 Volts per cell.~~

~~The "site-specific Vital DC busses" are the DC busses that provide monitoring and control capabilities for SAFETY SYSTEMS.~~

~~ECL Assignment Attributes: 3.1.3.B~~

SG1

ECL: General Emergency

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~Example~~ **Emergency Action Levels:**

Note: The Emergency Director should declare the General Emergency promptly upon determining that ~~(site-specific hours)~~4 hours has been exceeded, or will likely be exceeded.

SG1.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to ~~(site-specific emergency buses)~~1(2)-A-05 and 1(2)-A-06.

AND

b. **EITHER** of the following:

- Restoration of at least one AC emergency bus in less than ~~(site-specific hours)~~4 hours is not likely.
- ~~(Site-specific indication of an inability to adequately remove heat from the core)~~Conditions requiring entry into Core Cooling – Red Path (CSP-C.1) are met.

Definitions:

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

Basis:

For the purpose of classification under this EAL, evaluation of power sources should be made on each unit individually.

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

If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.

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

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station

blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

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

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

~~**Developer Notes:**~~

~~Although this IC and EAL may be viewed as redundant to the Fission Product Barrier ICs, it is included to provide for a more timely escalation of the emergency classification level.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The "site-specific hours" to restore AC power to an emergency bus should be based on the station blackout coping analysis performed in accordance with 10 CFR § 50.63 and Regulatory Guide 1.155, *Station Blackout*.~~

~~Site-specific indication of an inability to adequately remove heat from the core:~~

~~[*BWR*] Reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (as described in the EOP bases).~~

~~[*PWR*] Insert site-specific values for an incore/core exit thermocouple temperature and/or reactor vessel water level that drive entry into a core cooling restoration procedure (or otherwise requires implementation of prompt restoration actions). Alternately, a site may use incore/core exit thermocouple temperatures greater than 1,200°F and/or a reactor vessel water level that corresponds to approximately the middle of active fuel. Plants with reactor vessel level instrumentation that cannot measure down to approximately the middle of active fuel should use the lowest on-scale reading that is not above the top of active fuel. If the lowest on-scale reading is above the top of active fuel, then a reactor vessel level value should not be included.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters used in the Core Cooling Red Path.~~

~~ECL Assignment Attributes: 3.1.4.B~~

~~SG8~~SG2

ECL: General Emergency

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3, 4

~~**Example**~~ **Emergency Action Levels:**

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

SG2.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to ~~(site-specific emergency buses)~~1(2)-A-05 and 1(2)-A-06 for 15 minutes or longer.

AND

b. Indicated voltage is less than ~~(site-specific bus voltage value)~~115 VDC on **ALL** ~~(site-specific~~ Vital DC busses~~)~~ D-01, D-02, D--03 and D-04 for 15 minutes or longer.

Definitions:

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

Basis:

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

If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.

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

~~**Developer Notes:**~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The "site-specific bus voltage value" should be based on the minimum bus voltage necessary for adequate operation of SAFETY SYSTEM equipment. This voltage value should~~

~~incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed.~~

~~The typical value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries, the cell voltage is approximately 1.75 Volts per cell. For a 58 string battery set, the minimum voltage is approximately 1.81 Volts per cell.~~

~~The "site-specific Vital DC busses" are the DC busses that provide monitoring and control capabilities for SAFETY SYSTEMS.~~

~~This IC and EAL were added to Revision 6 to address operating experience from the March, 2011 accident at Fukushima Daiichi.~~

~~ECL Assignment Attributes: 3.1.4.B~~

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC .... Alternating Current
AOP .... Abnormal Operating Procedure
~~........ A~~
~~PRM .... Average Power Range Meter~~
ATWS .... Anticipated Transient Without Scram
~~........ B~~
~~&W .... Babcock and Wilcox~~
~~........ B~~
~~IIT .... Boron Injection Initiation Temperature~~
~~........ B~~
~~WR .... Boiling Water Reactor~~
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
~~CTMT/~~CNMT .... Containment
CSF .... Critical Safety Function
.... C
SFST .... Critical Safety Function Status Tree
~~DBA .... Design Basis Accident~~
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
~~........ E~~
~~PIP .... Emergency Plan Implementing Procedure~~
~~........ E~~
~~PR .... Evolutionary Power Reactor~~
~~........ E~~
~~PRI .... Electric Power Research Institute~~
~~........ E~~
~~RG .... Emergency Response Guideline~~
.... F
EMA .... Federal Emergency Management Agency
~~FSAR .... Final Safety Analysis Report~~
GE .... General Emergency
~~........ H~~
~~CTL .... Heat Capacity Temperature Limit~~
~~........ H~~
~~PCI .... High Pressure Coolant Injection~~
~~........ H~~
~~SI .... Human System Interface~~

IC .... Initiating Condition
~~..........~~I
D .... Inside Diameter
~~IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)~~
ISFSI .... Independent Spent Fuel Storage Installation
Keff .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
~~..........~~L
OCA .... Loss of Coolant Accident
~~.......... M~~
~~CR .... Main Control Room~~
~~.......... M~~
~~SIV .... Main Steam Isolation Valve~~
~~MSL .... Main Steam Line~~
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW .... Megawatt
NEI .... Nuclear Energy Institute
~~.......... N~~
~~PP .... Nuclear Power Plant~~
~~..........~~N
RC .... Nuclear Regulatory Commission
~~NSSS .... Nuclear Steam Supply System~~
~~..........~~N
ORAD .... North American Aerospace Defense Command
~~(NO)UE .... (Notification Of) Unusual Event~~
NUMARC[1] .... Nuclear Management and Resources Council
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
~~..........~~O
DCM~~/ODAM~~ .... Offsite Dose Calculation ~~(Assessment)~~ Manual
~~ORO .... Off-site Response Organization~~
PA .... Protected Area
~~.......... P~~
~~ACS .... Priority Actuation and Control System~~
PAG .... Protective Action Guideline
~~.......... P~~
~~ICS .... Process Information and Control System~~
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
~~.......... P~~
~~S .... Protection System~~
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen

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

~~...R~~
~~CC..Reactor Control Console~~
~~...R~~
~~CIC..Reactor Core Isolation Cooling~~
RCS..Reactor Coolant System
Rem, rem, REM ....Roentgen Equivalent Man
~~...R~~
~~ETS..Radiological Effluent Technical Specifications~~
RPS..Reactor Protection System
RPV..Reactor Pressure Vessel
~~...~~R
VLIS..Reactor Vessel Level Instrumentation System
~~...R~~
~~WCU..Reactor Water Cleanup~~
~~...S~~
~~AR..Safety Analysis Report~~
~~...S~~
~~AS..Safety Automation System~~
~~SBO..Station Blackout~~
SCBA..Self-Contained Breathing Apparatus
SG..Steam Generator
SI..Safety Injection
~~...S~~
~~ICS..Safety Information and Control System~~
~~...~~S
PDS..Safety Parameter Display System
~~SRO..Senior Reactor Operator~~
TEDE..Total Effective Dose Equivalent
~~...T~~
~~OAF..Top of Active Fuel~~
TSC..Technical Support Center
UE..Unusual Event
~~...~~
UFSAR..Updated Final Safety Analysis Report
WOG..Westinghouse Owners Group

APPENDIX B – DEFINITIONS

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

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

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

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

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

The following are key terms necessary for overall understanding the NEI 99-01 emergency classification scheme.

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

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

- ~~Notification of~~ Unusual Event (~~NO~~UE)
- Alert
- Site Area Emergency (SAE)

[+] ~~This term is sometimes shortened to Unusual Event (UE) or other similar site-specific terminology.~~

- General Emergency (GE)

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

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

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

CONFINEMENT BOUNDARY:~~ (Insert a site-specific definition for this term.) **Developer Note**~~ – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

CONTAINMENT CLOSURE:~~ (Insert a site-specific definition for this term.) **Developer Note** –~~ The procedurally defined conditions or actions taken to secure containment ~~(primary or secondary for BWR)~~ and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

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

FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.~~ **Developer Note** – This term is applicable to PWRs only.~~

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

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

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

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

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

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

~~NORMAL LEVELS: As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.~~

OWNER CONTROLLED AREA:~~(Insert a site-specific definition for this term.) **Developer Note**~~ This term is typically taken to mean the site property owned by~~,~~ or otherwise under the control of~~,~~ the licensee.~~In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan.~~

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

PROTECTED AREA: ~~(Insert a site-specific definition for this term.) **Developer Note** This term is typically taken to mean t~~The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY: ~~(Insert a site-specific definition for this term.) **Developer Note** This description should include all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.~~The reactor refueling cavity, spent fuel pool and fuel transfer canal.

RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.~~**Developer Note** This term is applicable to PWRs only.~~

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These ~~are typically~~ systems are classified as safety-related.~~**Developer Note** This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.~~

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

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

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

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

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

~~APPENDIX C – PERMANENTLY DEFUELED STATION ICs/EALs~~

~~Recognition Category PD provides a stand-alone set of ICs/EALs for a Permanently Defueled nuclear power plant to consider for use in developing a site-specific emergency classification scheme. For development, it was assumed that the plant had operated under a 10 CFR § 50 license and that the operating company has permanently ceased plant operations. Further, the company intends to store the spent fuel within the plant for some period of time.~~

~~When in a permanently defueled condition, the plant licensee typically receives approval from the NRC for exemption from specific emergency planning requirements. These exemptions reflect the lowered radiological source term and risks associated with spent fuel pool storage relative to reactor at-power operation. Source terms and accident analyses associated with plausible accidents are documented in the station's Final Safety Analysis Report (FSAR), as updated. As a result, each licensee will need to develop a site-specific emergency classification scheme using the NRC-approved exemptions, revised source terms, and revised accident analyses as documented in the station's FSAR.~~

~~Recognition Category PD uses the same ECLs as operating reactors; however, the source term and accident analyses typically limit the ECLs to an Unusual Event and Alert. The Unusual Event ICs provide for an increased awareness of abnormal conditions while the Alert ICs are specific to actual or potential impacts to spent fuel. The source terms and release motive forces associated with a permanently defueled plant would not be sufficient to require declaration of a Site Area Emergency or General Emergency.~~

~~A permanently defueled station is essentially a spent fuel storage facility with the spent fuel is stored in a pool of water that serves as both a cooling medium (i.e., removal of decay heat) and shield from direct radiation. These primary functions of the spent fuel storage pool are the focus of the Recognition Category PD ICs and EALs. Radiological effluent IC and EALs were included to provide a basis for classifying events that cannot be readily classified based on an observable events or plant conditions alone.~~

~~Appropriate ICs and EALs from Recognition Categories A, C, F, H, and S were modified and included in Recognition Category PD to address a spectrum of the events that may affect a spent fuel pool. The Recognition Category PD ICs and EALs reflect the relevant guidance in Section 3 of this document (e.g., the importance of avoiding both over-classification and under-classification). Nonetheless, each licensee will need to develop their emergency classification scheme using the NRC-approved exemptions, and the source terms and accident analyses specific to the licensee. Security-related events will also need to be considered.~~

~~Table PD-1: Recognition Category "PD" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~
~~**PD-AU1** Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-AA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.~~ ~~*Op. Modes: Not Applicable*~~
~~**PD-AU2** UNPLANNED rise in plant radiation levels.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-AA2** UNPLANNED rise in plant radiation levels that impedes plant access required to maintain spent fuel integrity.~~ ~~*Op. Modes: Not Applica*~~ble
~~**PD-SU1** UNPLANNED spent fuel pool temperature rise.~~ ~~*Op. Modes: Not Applicable*~~	
~~**PD-HU1** Confirmed SECURITY CONDITION or threat.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.~~ ~~*Op. Modes: Not Applicable*~~
~~**PD-HU2** Hazardous event affecting SAFETY SYSTEM equipment necessary for spent fuel cooling.~~ ~~*Op.*~~ Modes~~*: Not Applicable*~~	
~~**PD-HU3** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-HA3** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.~~ ~~*Op. Modes: Not Applicable*~~

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required~~.

~~PD-AU1~~

~~**ECL:** Notification of Unusual Event~~

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

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2)~~

~~**Notes:**~~

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

~~(1) Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.~~

~~(2) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.~~

~~**Basis:**~~

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

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

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

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

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

EAL #1 - This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL will typically be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

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

Escalation of the emergency classification level would be via IC PD-AA1.

Developer Notes:

The "site-specific effluent release controlling document" is the Radiological Effluent Technical Specifications (RETS) or, for plants that have implemented Generic Letter 89-01[11], the Offsite Dose Calculation Manual (ODCM). These documents implement regulations related to effluent controls (e.g., 10 CFR Part 20 and 10 CFR Part 50, Appendix I). As appropriate, the RETS or ODCM methodology should be used for establishing the monitor thresholds for this IC.

Listed monitors should include the effluent monitors described in the RETS or ODCM.

Developers may also consider including installed monitors associated with other potential effluent pathways that are not described in the RETS or ODCM[12][13]. If included, EAL values for these monitors should be determined using the most applicable dose/release limits presented in the RETS or ODCM. It is recognized that a calculated EAL value may be below what the monitor can read; in that case, the monitor does not need to be included in the list. Also, some monitors may not be governed by Technical Specifications or other license-related related requirements; therefore, it is important that the associated EAL and basis section clearly identify any limitations on the use or availability of these monitors.

---

[11] *Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program*

[12] This includes consideration of the effluent monitors described in the site emergency plan section(s) which address the requirements of 10 CFR 50.47(b)(8) and (9).

[13] Developers should keep in mind the requirements of 10 CFR 50.54(q) and the guidance provided by INPO related to emergency response equipment when considering the addition of other effluent monitors.

~~Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.~~

~~Radiation monitor readings should reflect values that correspond to a radiological release exceeding 2 times a release control limit. The controlling document typically describes methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values. In cases where a methodology is not adequately defined, developers should determine values consistent with effluent control regulations (e.g., 10 CFR Part 20 and 10 CFR Part 50 Appendix I) and related guidance.~~

~~For EAL #1 – Values in this EAL should be 2 times the setpoint established by the radioactivity discharge permit to warn of a release that is not in compliance with the specified limits. Indexing the value in this manner ensures consistency between the EAL and the setpoint established by a specific discharge permit.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.1.B~~

~~PD-AU2~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** UNPLANNED rise in plant radiation levels.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2)~~

~~(1) a. UNPLANNED water level drop in the spent fuel pool as indicated by ANY of the following:~~

~~(site-specific level indications).~~

~~**AND**~~

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

~~(site-specific list of area radiation monitors).~~

~~(2) Area radiation monitor reading or survey result indicates an UNPLANNED rise of 25 mR/hr over NORMAL LEVELS.~~

~~**Basis:**~~

~~This IC addresses elevated plant radiation levels caused by a decrease in water level above irradiated (spent) fuel or other UNPLANNED events. The increased radiation levels are indicative of a minor loss in the ability to control radiation levels within the plant or radioactive materials. Either condition is a potential degradation in the level of safety of the plant.~~

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

~~The effects of planned evolutions should be considered. Note that EAL #1 is applicable only in cases where the elevated reading is due to an UNPLANNED water level drop. EAL #2 excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials.~~

~~Escalation of the emergency classification level would be via IC PD-AA1 or PD-AA2.~~

~~**Developer Notes:**~~

~~For EAL #1 - Site-specific indications may include instrumentation values such as water level and area radiation monitor readings, and personnel reports. If available, video cameras may allow for remote observation. Depending on available instrumentation, the declaration may also be based on indications of water makeup rate and/or decreases in the level of a water storage tank.~~

~~For EAL #2 - The specified value of 25 mR/hr may be set to another value for a specific application with appropriate justification.~~

~~ECL Assignment Attributes: 3.1.1.B~~

~~PD-SU1~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** UNPLANNED spent fuel pool temperature rise.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:**~~

~~(1) UNPLANNED spent fuel pool temperature rise to greater than (site-specific °F).~~

~~**Basis:**~~

~~This IC addresses a condition that is a precursor to a more serious event and represents a potential degradation in the level of safety of the plant. If uncorrected, boiling in the pool will occur, and result in a loss of pool level and increased radiation levels.~~

~~Escalation of the emergency classification level would be via IC PD-AA1 or PD-AA2.~~

~~**Developer Notes:**~~

~~The site-specific temperature should be chosen based on the starting point for fuel damage calculations in the SAR. Typically, this temperature is 125° to 150°F. Spent Fuel Pool temperature is normally maintained well below this point thus allowing time to correct the cooling system malfunction prior to classification.~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~PD-HU1~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Confirmed SECURITY CONDITION or threat.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).~~

~~(2) Notification of a credible security threat directed at the site.~~

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

~~**Basis:**~~

~~This IC addresses events that pose a threat to plant personnel or the equipment necessary to maintain cooling of spent fuel, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under IC PD-HA1.~~

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

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

~~EAL #1 references (site-specific security shift supervision) because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.~~

~~EAL #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with (site-specific procedure).~~

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

~~Emergency plans and implementing procedures are public documents; therefore, EALs should~~

~~not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

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

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~The (site-specific procedure) is the procedure(s) used by Control Room and/or Security personnel to determine if a security threat is credible, and to validate receipt of aircraft threat information.~~

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

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~PD-HU2~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Hazardous event affecting SAFETY SYSTEM equipment necessary for spent fuel cooling.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:**~~

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

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

~~**AND**~~

~~b. The event has damaged at least one train of a SAFETY SYSTEM needed for spent fuel cooling.~~

~~**AND**~~

~~c. The damaged SAFETY SYSTEM train(s) cannot, or potentially cannot, perform its design function based on **EITHER**:~~

- ~~Indications of degraded performance~~
- ~~VISIBLE DAMAGE~~

~~**Basis:**~~

~~This IC addresses a hazardous event that causes damage to at least one train of a SAFETY SYSTEM needed for spent fuel cooling. The damage must be of sufficient magnitude that the system(s) train cannot, or potentially cannot, perform its design function. This condition reduces the margin to a loss or potential loss of the fuel clad barrier, and therefore represents a potential degradation of the level of safety of the plant.~~

~~For EAL 1.c, the first bullet addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available.~~

~~For EAL 1.c, the second bullet addresses damage to a SAFETY SYSTEM train that is not in service/operation or readily apparent through indications alone. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.~~

~~Escalation of the emergency classification level could, depending upon the event, be based on any of the Alert ICs; PD-AA1, PD-AA2, PD-HA1 or PD-HA3.~~

~~**Developer Notes:**~~

~~For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).~~

~~Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1C~~

~~PD-HU3~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:**~~

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

~~**Basis:**~~

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

~~PD-AA1~~

~~**ECL:** Alert~~

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

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2 or 3 or 4)~~

~~**Notes:**~~

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

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

~~(site-specific monitor list and threshold values)~~

~~(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).~~

~~(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.~~

~~(4) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point):~~

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

~~**Basis:**~~

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

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

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

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

~~Developer Notes:~~

~~While this IC may not be met absent challenges to the cooling of spent fuel, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant conditions alone.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE….".~~

~~The EPA PAG guidance provides for the use adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous and liquid effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 10 mrem TEDE or 50 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or~~

~~atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for IC PD-AU1.~~

- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for IC PD-AU1.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~PD-AA2~~

~~**ECL:** Alert~~

~~**Initiating Condition:** UNPLANNED rise in plant radiation levels that impedes plant access required to maintain spent fuel integrity.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2)~~

~~(1) UNPLANNED dose rate greater than 15 mR/hr in **ANY** of the following areas requiring continuous occupancy to maintain control of radioactive material or operation of systems needed to maintain spent fuel integrity:~~

~~(site-specific area list)~~

~~(2) UNPLANNED Area Radiation Monitor readings or survey results indicate a rise by 100 mR/hr over NORMAL LEVELS that impedes access to **ANY** of the following areas needed to maintain control of radioactive material or operation of systems needed to maintain spent fuel integrity.~~

~~(site-specific area list)~~

~~**Basis:**~~

~~This IC addresses increased radiation levels that impede necessary access to areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain systems needed to maintain spent fuel integrity. As used here, 'impede' includes hindering or interfering, provided that the interference or delay is sufficient to significantly threaten necessary plant access. It is this impaired access that results in the actual or potential substantial degradation of the level of safety of the plant.~~

~~This IC does not apply to anticipated temporary increases due to planned events.~~

~~**Developer Notes:**~~

~~The value of 15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, *Clarification of TMI Action Plan Requirements*, provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.~~

~~The specified value of 100 mR/hr may be set to another value for a specific application with appropriate justification.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~PD-HA1~~

~~**ECL:** Alert~~

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

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2)~~

~~(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).~~

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

~~**Basis:**~~

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

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

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

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

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

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

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

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

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

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

~~Developer Notes:~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

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

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the OWNER CONTROLLED AREA.~~

~~ECL Assignment Attributes: 3.1.2.D~~

~~PD-HA3~~

~~ECL:~~ ~~Alert~~

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

~~Operating Mode Applicability:~~ ~~Not Applicable~~

~~Example Emergency Action Levels:~~

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

~~Basis:~~

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

ATTACHMENT 2

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 286, ADOPTION OF EMERGENCY ACTION LEVEL SCHEME PURSUANT TO NEI 99-01 REVISION 6, "DEVELOPMENT OF EMERGENCY ACTION LEVELS FOR NON-PASSIVE REACTORS"

UPDATED CLEAN COPY OF NEI 99-01 REVISION 6

136 pages follow

Point Beach Emergency Action Levels Bases Document

TBD, 2018

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TABLE OF CONTENTS

POINT BEACH EMERGENCY ACTION LEVELS BASIS DOCUMENT

1 REGULATORY BACKGROUND

1.1 OPERATING REACTORS

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

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

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

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

1.2 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

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

The generic ICs and EALs for an ISFSI are presented in Section 8, ISFSI ICs/EALs. IC EU1 covers the spectrum of credible natural and man-made events included within the scope of an ISFSI design. This IC is not applicable to installations or facilities that may process and/or repackage spent fuel (e.g., a Monitored Retrievable Storage Facility or an ISFSI at a spent fuel processing facility). In addition, appropriate aspects of IC HU1 and IC HA1 should also be included to address a HOSTILE ACTION directed against an ISFSI.

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

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

1.3 NRC ORDER EA-12-051

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

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

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

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

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

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

2 KEY TERMINOLOGY USED IN NEI 99-01

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

<table>
<tr><td colspan="4">Emergency Classification Level</td></tr>
<tr><td>Unusual Event</td><td>Alert</td><td>SAE</td><td>GE</td></tr>
<tr><td>↓</td><td>↓</td><td>↓</td><td>↓</td></tr>
<tr><td>Initiating Condition</td><td>Initiating Condition</td><td>Initiating Condition</td><td>Initiating Condition</td></tr>
<tr><td>↓</td><td>↓</td><td>↓</td><td>↓</td></tr>
<tr><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td></tr>
<tr><td colspan="4">(1) - When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition. This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.</td></tr>
</table>

2.1 EMERGENCY CLASSIFICATION LEVEL (ECL)

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

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

2.1.1 Unusual Event (UE)

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

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

2.1.2 Alert

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

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

2.1.3 Site Area Emergency

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

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

2.1.4 General Emergency (GE)

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

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

2.2 INITIATING CONDITION (IC)

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

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

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

2.3 EMERGENCY ACTION LEVEL (EAL)

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

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

2.4 FISSION PRODUCT BARRIER THRESHOLD

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

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

- Fuel Clad
- Reactor Coolant System (RCS)
- Containment

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

3 DESIGN OF THE PBNP EMERGENCY CLASSIFICATION SCHEME

3.1 ASSIGNMENT OF EMERGENCY CLASSIFICATION LEVELS (ECLS)

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

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

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

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

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

3.1.1 Unusual Event (UE)

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

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

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

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

3.1.2 Alert

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

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

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

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

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

3.1.3 Site Area Emergency (SAE)

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

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

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

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

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

3.1.4 General Emergency (GE)

A General Emergency, as defined in section 2.1.4, includes but is not limited to an event

or condition that involves:

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

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

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

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

3.1.5 Risk-Informed Insights

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

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

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

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

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

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

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

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

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

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

3.3 PBNP-SPECIFIC ORGANIZATION AND PRESENTATION OF GENERIC INFORMATION

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

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

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

The following information and guidance is provided for each IC:

- **ECL** – the assigned emergency classification level for the IC.

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

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

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

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

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

3.4 IC AND EAL MODE APPLICABILITY

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

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

MODE APPLICABILITY MATRIX

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

PBNP OPERATING MODES

MODE	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥0.99	> 5	NA
2	Startup	≥0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥350
4	Hot Shutdown[(b)]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown [(b)]	< 0.99	NA	≤ 200
6	Refueling [(c)]	NA	NA	NA
N/A	Defueled	All fuel removed from the reactor vessel (full core offload during refueling or extended outage)		
	(a) Excluding decay heat (b) All reactor vessel head closure bolts fully tensioned. (c) One or more reactor vessel head closure bolts less than fully tensioned.			

4 PBNP SCHEME DEVELOPMENT

4.1 GENERAL DEVELOPMENT PROCESS

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

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

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

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

4.2 CRITICAL CHARACTERISTICS

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

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

4.3 INSTRUMENTATION USED FOR EALS

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

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

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

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

5 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

5.1 GENERAL CONSIDERATIONS

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

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

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

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

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

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

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

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

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

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

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.); the EAL and/or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

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

5.2 CLASSIFICATION METHODOLOGY

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

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

5.3 CLASSIFICATION OF MULTIPLE EVENTS AND CONDITIONS

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. For example:

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

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

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

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

5.4 CONSIDERATION OF MODE CHANGES DURING CLASSIFICATION

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

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

5.5 CLASSIFICATION OF IMMINENT CONDITIONS

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

5.6 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND DOWNGRADING

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

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

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

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

5.7 CLASSIFICATION OF SHORT-LIVED EVENTS

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

5.8 CLASSIFICATION OF TRANSIENT CONDITIONS

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

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

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

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

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

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

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

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

5.10 RETRACTION OF AN EMERGENCY DECLARATION

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

6 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

ECL: Unusual Event

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

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

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

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

Monitor	Reading
1(2)RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation	1.4E-2 μCi/cc
2RE-305 CTMNT Purge Exhaust Low Range Gas with both purge and GS building ventilation in operation	9.4E-3 μCi/cc
2RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation	9.4E-3 μCi/cc
2RE-307 CTMNT Purge Exhaust Mid-Range Gas with only GS building ventilation in operation	2.8E-2 μCi/cc
2RE-307 CTMNT Purge Exhaust Mid-Range Gas with only forced vent of containment	1.0E+1 μCi/cc
2RE-309 CTMNT Purge Exhaust High Range Gas with only forced vent of containment	1.0E+1 μCi/cc
RE-315 AB Exhaust Low Range Gas	5.4E-3 μCi/cc
RE-317 AB Exhaust Mid-Range Gas	5.4E-3 μCi/cc
RE-325 Drumming Area Exhaust Low Range Gas	8.4E-3 μCi/cc
RE-327 Drumming Area Exhaust Mid-Range Gas	8.4E-3 μCi/cc
1(2)RE-229 Service Water Overboard	2.3E-3 μCi/cc

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

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

Definitions:

None

Basis:

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

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

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

EAL RU1.1 - This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

EAL RU1.2 - This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL will typically be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

EAL RU1.3 - This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC RA1.

ECL: Unusual Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel.

Operating Mode Applicability: All

Emergency Action Levels:

RU2.1 a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:

- Spent fuel pool low water level alarm
- Visual observation

AND

b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors.

- RE-105 SFP Area Low Range Radiation Monitor
- RE-135 SFP Area High Range Radiation Monitor
- 1(2)RE-102 El. 66' CONTAINMENT Low Range Monitor

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool and fuel transfer canal.

Basis:

This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. The low level alarm is actuated by LC-634, SFP Level Indicator at 62 ft. 8 in. based on maintaining at least 6 ft. of water on a withdrawn fuel assembly. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

ECL: Alert

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RA1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RA1.1 Reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer:

Monitor	Reading
1(2)RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation	6.0E+0 µCi/cc
1(2)RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation	6.0E+0 µCi/cc
2RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation	4.0E+0 µCi/cc
2RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation	4.0E+0 µCi/cc
2RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation	1.2E+1 µCi/cc
RE-317 AB Exhaust Mid-Range Gas	1.0E+0 µCi/cc
RE-319 AB Exhaust High Range Gas	1.0E+0 µCi/cc
RE-327 Drumming Area Exhaust Mid-Range Gas	1.6E+0 µCi/cc

RA1.2 Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond SITE BOUNDARY.

RA1.3 Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.

RA1.4 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

This IC is modified by a note that EAL RA1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

For EAL RA1.3, there are no site-specific liquid radiation monitors capable of monitoring liquid effluent releases at the classification threshold for this EAL because their detector operating range is exceeded prior to reaching these levels. Entry into this EAL for a liquid radioactivity release will be based on sampling initiated due to entry into EAL RU1. In practical terms, this means that entry into IC RU1 will start sampling (per RMS Alarm Setpoint and Response Book) which will then allow detection of the setpoint for RA1.

Escalation of the emergency classification level would be via IC RS1.

ECL: Alert

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.

Operating Mode Applicability: All

Emergency Action Levels:

RA2.1 Uncovery of irradiated fuel in the REFUELING PATHWAY.

RA2.2 Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by a reading on **ANY** of the following radiation monitors greater than the value shown:

Monitor	Reading
RE-105 SFP Area Low Range Radiation Monitor	4 R/hr
1(2)RE-126 Containment High Radiation Monitor	7 R/hr
1(2)RE-127 Containment High Radiation Monitor	7 R/hr
1(2)RE-128 Containment High Radiation Monitor	7 R/hr

RA2.3 Lowering of spent fuel pool level to 49 ft.0 in.

Definitions:

REFUELING PATHWAY – The reactor refueling cavity, spent fuel pool and fuel transfer canal.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC EU1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

EAL RA2.1

This EAL escalates from RU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used. Classification of an event using this EAL should be based on the totality of available indications, reports, and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

EAL RA2.2

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

EAL RA2.3

Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the emergency classification level would be via ICs RS1 or RS2.

ECL: Alert

Initiating Condition: Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.

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.

RA3.1 Dose rate greater than 15 mR/hr in **ANY** of the following areas:

- Control Room (RE-101)
- Central Alarm Station **AND** Secondary Alarm Station (by survey)

RA3.2 An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas:

Area	Mode
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B32 MCC Area	4

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For EAL RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

An emergency declaration is not warranted if any of the following conditions apply.

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

The list of plant rooms or areas in EAL RA3.2 was generated from a step-by-step review of OP-3A, 3B, 3C, 5D, and 7A.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

ECL: Site Area Emergency

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RS1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RS1.1 Reading on **ANY** of the following radiation monitor greater than the reading shown for 15 minutes or longer:

Monitor	Reading
1(2)RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation	6.0E+1 µCi/cc
1(2)RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation	6.0E+1 µCi/cc
2RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation	4.0E+1 µCi/cc
2RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation	4.0E+1 µCi/cc
2RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation	1.2E+2 µCi/cc
RE-317 AB Exhaust Mid-Range Gas	1.0E+1 µCi/cc
RE-319 AB Exhaust High Range Gas	1.0E+1 µCi/cc
RE-327 Drumming Area Exhaust Mid-Range Gas	1.6E+1 µCi/cc

RS1.2 Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY.

RS1.3 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

This IC is modified by a note that EAL RS1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

RS2

ECL: Site Area Emergency

Initiating Condition: Spent fuel pool level at 40 ft. 8 in.

Operating Mode Applicability: All

Emergency Action Levels:

RS2.1 Lowering of spent fuel pool level to 40 ft. 8 in..

Definitions:

None

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

It is recognized that this IC would likely not be met until well after another Site Area Emergency IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC RG1 or RG2.

RG1

ECL: General Emergency

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the General Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RG1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RG1.1 Reading on **ANY** of the following radiation monitor greater than the reading shown for 15 minutes or longer:

Monitor	**Reading**
1(2)RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation	6.0E+2 μCi/cc
2RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation	4.0E+2 μCi/cc
2RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation	1.2E+3 μCi/cc
RE-317 AB Exhaust Mid-Range Gas	1.0E+2 μCi/cc
RE-319 AB Exhaust High Range Gas	1.0E+2 μCi/cc

RG1.2 Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY.

RG1.3 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

This IC is modified by a note that EAL RG1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

ECL: General Emergency

Initiating Condition: Spent fuel pool level cannot be restored to at least 40 ft.8 in. for 60 minutes or longer.

Operating Mode Applicability: All

Emergency Action Levels:

Note: The Emergency Director should declare the General Emergency promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.

RG2.1 Spent fuel pool level cannot be restored to at least 40 ft. 8 in. for 60 minutes or longer.

Definitions:

None

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another General Emergency IC was met; however, it is included to provide classification diversity.

7 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

ECL: Unusual Event

Initiating Condition: UNPLANNED loss of reactor vessel/RCS inventory for 15 minutes or longer.

Operating Mode Applicability: 5, 6

Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU1.1 UNPLANNED loss of reactor coolant results in reactor vessel/RCS level less than a required lower limit for 15 minutes or longer.

CU1.2 a. Reactor vessel/RCS level cannot be monitored.

AND

b. UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor reactor vessel/RCS level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

EAL CU1.1 recognizes that the minimum required reactor vessel/RCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

EAL CU1.2 addresses a condition where all means to determine reactor vessel/RCS level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the reactor vessel/RCS.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

ECL: 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:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU2.1 a. AC power capability to 1(2)A-05 and 1(2)A-06 is reduced to a single power source for 15 minutes or longer.

AND

b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

Note: with respect to this EAL, "Station Blackout is Unit 1(2) specific."

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.

- Normal Unit 1(2) offsite power sources include:
  - 345 KVAC 1(2)X-03 through the 13.8 KVAC system to the LVSAT 1(2)X-04
  - 345 KVAC backfed through the 19 KVAC system to the UAT 1(2)X-02

- Normal Unit 1(2) onsite power sources consist of:
  - emergency diesel generators
  - gas turbine generator
  - unit main turbine generator
  - power supplied from the opposite unit

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

An "AC power source" is a source recognized in AOPs and EOPs (including Beyond Design Basis event procedures), and capable of supplying required power to an emergency bus. Some examples of this Initiating Condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from an offsite power source.

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

ECL: Unusual Event

Initiating Condition: UNPLANNED increase in RCS temperature.

Operating Mode Applicability: 5, 6

Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU3.1 UNPLANNED increase in RCS temperature to greater than 200°F.

CU3.2 Loss of **ALL** RCS temperature and reactor vessel/RCS level indication for 15 minutes or longer.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

CONTAINMENT CLOSURE: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Basis:

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or the inability to determine RCS temperature and level, represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

EAL CU3.1 involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

EAL CU3.2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

ECL: Unusual Event

Initiating Condition: Loss of Vital DC power for 15 minutes or longer.

Operating Mode Applicability: 5, 6

Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU4.1 Indicated voltage is less than 115 VDC on required Vital DC buses D-01, D-02, D-03, or D-04 for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the Vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

The safety-related 125 VDC system consist of four main buses; D-01, D-02, D-03, and D-04.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category R.

ECL: Unusual Event

Initiating Condition: Loss of all onsite or offsite communications capabilities.

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Levels:

CU5.1 Loss of **ALL** of the following onsite communication methods:

- Plant Public Address System (Gai-Tronics)
- Commercial Phones
- PBX Phones
- Security Radio
- Portable Radios

CU5.2 Loss of ALL of the following offsite response organization communications methods:

- Nuclear Accident Reporting System (NARS)
- Commercial Phones
- PBX Phones
- Satellite Phones
- Manitowoc County Sheriff's Department Radio

CU5.3 Loss of **ALL** of the following NRC communications methods:

- FTS Phone System
- Commercial Phones
- PBX Phones
- Satellite Phones

Definitions:

None

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to offsite response organizations and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL CU5.1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL CU5.2 addresses a total loss of the communications methods used to notify all offsite response organizations of an emergency declaration. The offsite response organizations referred to here are the State of Wisconsin, Manitowoc County, and Kewaunee County.

EAL CU5.3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

ECL: Alert

Initiating Condition: Loss of reactor vessel/RCS inventory.

Operating Mode Applicability: 5, 6

Emergency Action Levels:

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CA1.1 Loss of reactor vessel/RCS inventory as indicated by level less than 16% on LI-447 / LI-447A.

CA1.2 a. Reactor vessel/RCS level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels due to a loss of reactor vessel/RCS inventory.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For EAL CA1.1, a lowering of water level below 16% on LI-447 / LI-447A indicates that operator actions have not been successful in restoring and maintaining reactor vessel/RCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery. The LI-447/LI-447A threshold corresponds to the minimum shutdown reactor vessel level required for operation of RHR without air binding the suction.

Although related, EAL CA1.1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

For EAL CA1.2, the inability to monitor reactor vessel/RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the reactor vessel/RCS.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1

If the reactor vessel/RCS inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

ECL: Alert

Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Levels:

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CA2.1 Loss of **ALL** offsite and **ALL** onsite AC Power to 1(2)A-05 and 1(2)A-06 for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

For the purpose of classification under this EAL, evaluation of power sources should be made on each unit individually.

If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC CS1 or RS1.

ECL: Alert

Initiating Condition: Inability to maintain the plant in cold shutdown.

Operating Mode Applicability: 5, 6

Emergency Action Levels:

Note: The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CA3.1 UNPLANNED increase in RCS temperature to greater than 200°F for greater than the duration specified in the following table.

Table: RCS Heat-up Duration Thresholds		
RCS Status	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
Intact (but not at reduced inventory)	Not applicable	60 minutes*
Not intact (or at reduced inventory)	Established	20 minutes*
	Not Established	0 minutes
* If RHR is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

CA3.2 UNPLANNED RCS pressure increase greater than 25 psig. (This EAL does not apply during water-solid plant conditions.)

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

CONTAINMENT CLOSURE: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Basis:

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact, or RCS inventory is reduced (e.g., mid-loop operation in PWRs). The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact or is at reduced inventory, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because

1) the evaporated reactor coolant may be released directly into the Containment atmosphere and subsequently to the environment, and
2) there is reduced reactor coolant inventory above the top of irradiated fuel.

EAL CA3.2 provides a pressure-based indication of RCS heat-up.

Escalation of the emergency classification level would be via IC CS1 or RS1.

ECL: Alert

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

Operating Mode Applicability: 5, 6

Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

CA6.1 a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Lake level greater than or equal to 9.0 ft. (Plant elevation)
- Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

AND

b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Definitions:

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction, or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria CA6.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC RS1.

ECL: Site Area Emergency

Initiating Condition: Loss of reactor vessel/RCS inventory affecting core decay heat removal capability.

Operating Mode Applicability: 5, 6

Emergency Action Levels:

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

CS1.1 a. Reactor vessel/RCS level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **ANY** of the following:

- Containment High Radiation Monitor, 1(2)RE-126, 1(2)RE-127, or 1(2)RE-128, reading greater than 100 R/hr
- Erratic source range monitor indication
- UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses a significant and prolonged loss of reactor vessel/RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS/reactor vessel level cannot be restored, fuel damage is probable.

In EAL CS1.1.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

For EAL CS1.1.a, the calculated radiation level on the Containment High Radiation Monitors (RE-126, RE-127, or RE-128) is without the reactor head in place. Calculated radiation levels with the reactor head in place are below the usable scale of these monitors.

The inability to monitor reactor vessel/RCS level may be caused by instrumentation and/or power failures, equipment not calibrated for the plant conditions (e.g., hot cal only), or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the reactor vessel/RCS.

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

Escalation of the emergency classification level would be via IC CG1 or RG1.

ECL: General Emergency

Initiating Condition: Loss of reactor vessel/RCS inventory affecting fuel clad integrity with containment challenged.

Operating Mode Applicability: 5, 6

Emergency Action Levels:

Note: The Emergency Director should declare the General Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

CG1.1 a. Reactor vessel/RCS level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **ANY** of the following:

- Containment High Radiation Monitors, 1(2)RE-126, 1(2)RE-127, or 1(2)RE-128, reading greater than 100 R/hr
- Erratic source range monitor indication
- UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery

AND

c. **ANY** indication from the Containment Challenge Table C-1.

Containment Challenge Table C-1
■ CONTAINMENT CLOSURE not established*
■ 6% H^2 exists inside containment
■ UNPLANNED increase in containment pressure

* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute time limit, then declaration of a General Emergency is not required.

Definitions:

CONTAINMENT CLOSURE: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS/reactor vessel level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

In EAL CG1.1.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

For EAL CG1.1.b, the calculated radiation level on the Containment High Radiation Monitors, 1(2)RE-126, 1(2)RE-127, or 1(2)RE-128, is without the reactor head in place. Calculated radiation levels with the reactor head in place are below the usable scale of these monitors.

The inability to monitor reactor vessel/RCS level may be caused by instrumentation and/or power failures, equipment not calibrated for the plant conditions (e.g., hot cal only), or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the reactor vessel/RCS.

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*.

8 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

ISFSI MALFUNCTION

EU1

ECL: Unusual Event

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: All

Emergency Action Levels:

EU1.1 Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than the values shown below on the surface of the spent fuel cask.

32 PT DSC	
Front Surface	1700 mrem/hr
Door Centerline	400 mrem/hr
End Shield Wall Exterior	12 mrem/hr
VSC-24	
Sides	200 mrem/hr
Top	400 mrem/hr
Air Inlets	700 mrem/hr
Air Outlets	200 mrem/hr

Definition:

CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

Basis:

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

9 FISSION PRODUCT BARRIER ICS/EALS

Table 9-F-1: EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

FA1 ALERT	**FS1 SITE AREA EMERGENCY**	**FG1 GENERAL EMERGENCY**
Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.	Loss or Potential Loss of any two barriers.	Loss of any two barriers and Loss or Potential Loss of the third barrier.
Operating Mode Applicability: 1, 2, 3, 4	**Operating Mode Applicability:** 1, 2, 3, 4	**Operating Mode Applicability:** 1, 2, 3, 4

Fuel Clad Barrier		**RCS Barrier**		**Containment Barrier**	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
A. Conditions requiring entry into Core Cooling RED Path (CSP C.1) are met.	A. Conditions requiring entry into Core Cooling ORANGE Path (CSP C.2) are met. **OR** B. Conditions requiring entry into Heat Sink RED Path (CSP H.1) are met.	Not Applicable	A. Conditions requiring entry into Heat Sink RED Path (CSP H.1) are met. **OR** B. Conditions requiring entry into RCS Integrity RED Path (CSP P.1) are met.	Not Applicable	A. 1. Conditions requiring entry into Core Cooling RED Path (CSP C.1) are met. **AND** 2. CSP C.1 not effective within 15 minutes.
2. RCS or SG Tube Leakage		**2. RCS or SG Tube Leakage**		**2. RCS or SG Tube Leakage**	
Not Applicable	Not Applicable	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage.	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable

3. RCS Activity / Containment Radiation		3. RCS Activity / Containment Radiation		3. RCS Activity / Containment Radiation	
A. Containment radiation monitor reading greater than 577 R/hr indicated on **ANY** of the following. • 1(2)RE-126 • 1(2)RE-127 • 1(2)RE-128 **OR** B. 1(2)RE-109 greater than 4,500 mR/hr	Not Applicable	A. Containment radiation monitor reading greater than 11 R/hr indicated on **ANY** of the following: • 1(2)RE-126 • 1(2)RE-127 • 1(2)RE-128	Not Applicable	Not Applicable	A. Containment radiation monitor reading greater than 18,500 R/hr indicated on **ANY** of the following. • 1(2)RE-126 • 1(2)RE-127 • 1(2)RE-128
4. Containment Integrity or Bypass		**4. Containment Integrity or Bypass**		**4. Containment Integrity or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Containment isolation is required **AND EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment pressure greater than 60 psig. **OR** B. 6% H^2 inside containment. **OR** C. 1. Containment pressure greater than 25 psig. **AND** 2. Less than one full train of depressurization equipment is operating per design for 15 minutes or longer.

5. Emergency Director Judgment		5. Emergency Director Judgment		5. Emergency Director Judgment	
A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis Information For
Fission Product Barrier Table 9-F-1

FUEL CLAD BARRIER THRESHOLDS:

The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

1. Critical Safety Function Status

Loss 1.A

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier. CSP-C.1 is the Critical Safety Procedure that provides directions to restore core cooling.

Potential Loss 1.A

This reading indicates temperatures within the core are sufficient to allow the onset of heat-induced cladding damage.

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur. CSP-C.2 is the Critical Safety Procedure that provides directions to restore adequate core cooling.

Potential Loss 1.B

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. Verification of the EALs for CSP-H.1 should include an assessment of if feed flow was reduced based on the actions performed for an uncontrolled depressurization of both steam generators. If this is the case, then declaration requirements should not be considered to be met. This does not affect the time that the CSFST initially changed color and met the CSP entry conditions for potential EAL event classification. Alternatively, if CSP-H.1 was entered during a loss of coolant accident, it may be exited if secondary heat sink is not required based on RCS pressure less than non-faulted S/G pressure. If this is the case, then declaration requirements should not be considered to be met.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to RCS Barrier Potential Loss threshold 2.A; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using this threshold is not warranted.

FUEL CLAD BARRIER THRESHOLDS:

2. RCS or SG Tube Leakage

There are no Loss or Potential Loss thresholds associated with RCS or SG Tube Leakage.

3. RCS Activity / Containment Radiation

Loss 3.A

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold 3.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

Loss 3.B

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

5. Emergency Director Judgment

Loss 5.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

Potential Loss 5.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

RCS BARRIER THRESHOLDS:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

1. Critical Safety Function Status

There is no Loss threshold associated with Critical Safety Function Status.

Potential Loss 1.A

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. If CSP-H.1 was entered during a loss of coolant accident, it may be exited if secondary heat sink is not required based on RCS pressure less than non-faulted S/G pressure. If this is the case, then declaration requirements should not be considered to be met.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold 1.B; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

CSP-H.1 is the Critical Safety Procedure that provides directions if the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a Potential Loss of the RCS barrier.

In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using this threshold is not warranted.

Potential Loss 1.B

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

CSP-P.1 is the Critical Safety Procedure that provides directions to avoid, or limit, thermal shock or pressurized thermal shock to the reactor pressure vessel or overpressurization conditions at low temperatures.

IF CSP-P.1 is entered during a large break loss of coolant accident, it may be exited if RCS pressure is low enough to allow for RHR forward flow. If CSP-P.1 is exited for this reason, then the Potential Loss criteria should not be considered met.

RCS BARRIER THRESHOLDS:

2. RCS or SG Tube Leakage

Loss 2.A

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

Potential Loss 2.A

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

3. RCS Activity / Containment Radiation

Loss 3.A

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold 3.A since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

RCS BARRIER THRESHOLDS:

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

5. Emergency Director Judgment

Loss 5.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.

Potential Loss 5.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CONTAINMENT BARRIER THRESHOLDS:

The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

1. Critical Safety Function Status Tree

There is no Loss threshold associated with CSFST.

Potential Loss 1.A

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

2. RCS or SG Tube Leakage

Loss 2.A

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss 1.A and Loss 1.A, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably [*part of the FAULTED definition*] and the faulted steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

CONTAINMENT BARRIER THRESHOLDS:

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG atmospheric dump valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The emergency classification levels resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU4	Unusual Event per SU4
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1	Alert per FA1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1	Alert per FA1

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

CONTAINMENT BARRIER THRESHOLDS:

3. RCS Activity / Containment Radiation

There is no Loss threshold associated with RCS Activity / Containment Radiation.

Potential Loss 3.A

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.

4. Containment Integrity or Bypass

Loss 4.A

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both thresholds 4.A.1 and 4.A.2.

4.A.1 – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage (L_a)). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Director will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 9-F-2. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

CONTAINMENT BARRIER THRESHOLDS:

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

4.A.2 – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 9-F-2. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 9-F-2. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump or system piping developed a leak that allowed steam/water to enter the Auxiliary Building, then threshold 4.B would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

CONTAINMENT BARRIER THRESHOLDS:

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to a closed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold 2.A.

Loss 4.B

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure 9-F-2. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold 2.A to be met.

Potential Loss 4.A

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

CONTAINMENT BARRIER THRESHOLDS:

Potential Loss 4.B

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.

Potential Loss 4.C

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, containment accident fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

During a design basis accident, a minimum of two CONTAINMENT Accident Fan Cooler Units with their accident fans running and one CONTAINMENT spray train are required to maintain the CONTAINMENT peak pressure and temperature below the design limits. Each CONTAINMENT Spray train is a CONTAINMENT spray pump, spray header, nozzles, valves and piping. Each CONTAINMENT Accident Fan Cooler Unit consists of cooling coils, accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path.

5. Emergency Director Judgment

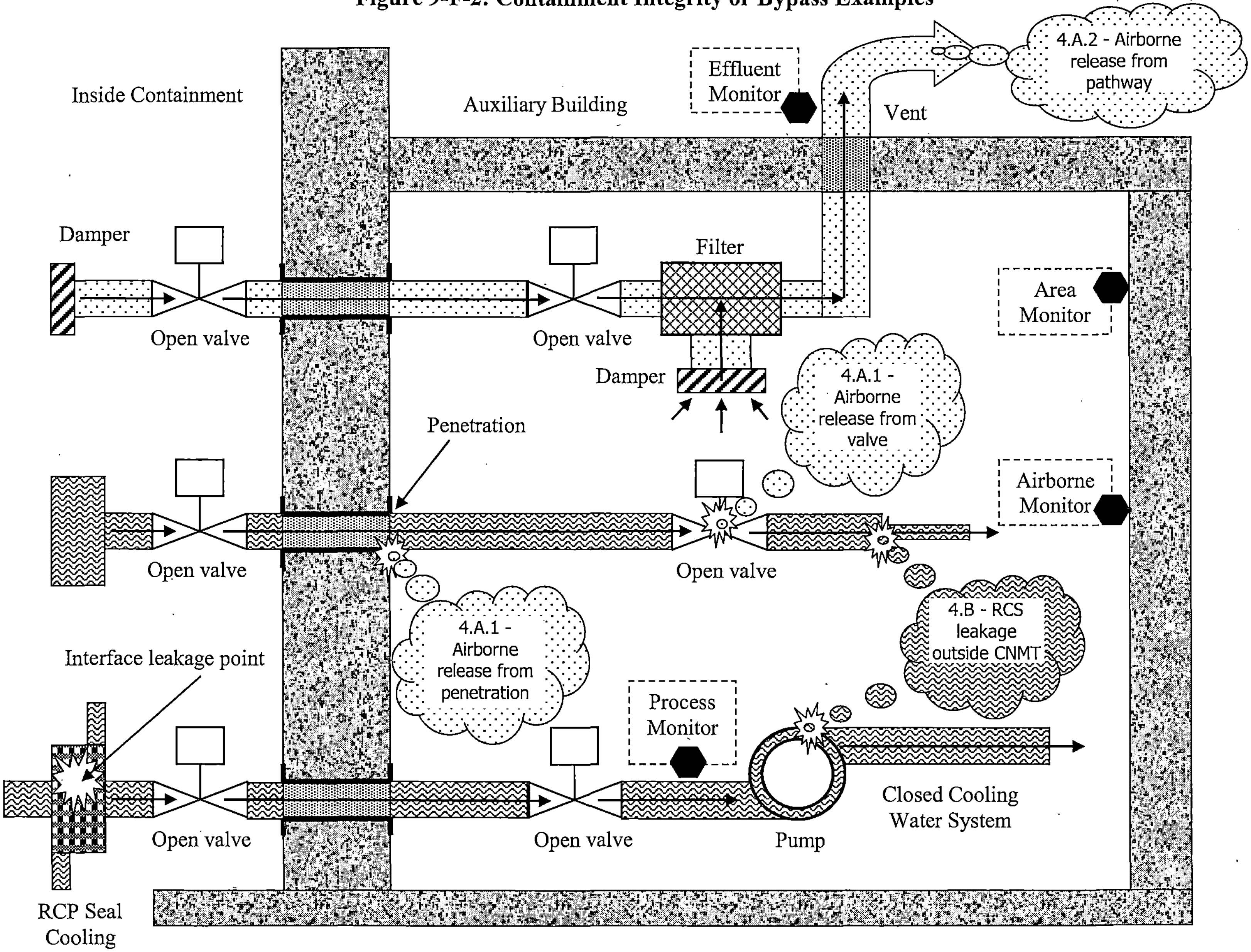
Loss 5.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

Potential Loss 5.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Figure 9-F-2: Containment Integrity or Bypass Examples



10 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

ECL: Unusual Event

Initiating Condition: Confirmed SECURITY CONDITION or threat.

Operating Mode Applicability: All

Emergency Action Levels:

HU1.1 A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the Security Shift Supervisor.

HU1.2 Notification of a credible security threat directed at PBNP.

HU1.3 A validated notification from the NRC providing information of an aircraft threat.

Definitions:

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

HOSTILE ACTION: An act toward PBNP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR 73.71 or 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervisor and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and offsite response organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

EAL HU1.1 references Security Shift Supervisor because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.390 information.

EAL HU1.2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with SY-AA-102-1014, Threat Assessment and Reporting.

EAL HU1.3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with AOP-29, Security Threat.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.

Escalation of the emergency classification level would be via IC HA1.

ECL: Unusual Event

Initiating Condition: Seismic event greater than OBE levels.

Operating Mode Applicability: All

Emergency Action Levels:

HU2.1 Seismic event greater than Operating Basis Earthquake (OBE) as indicated by seismic monitor indication of ground acceleration greater than:

- 0.06 g horizontal

OR

- 0.04 g vertical.

Definitions:

None

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE)[1]. An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE)[2] should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event. The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

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[1] An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.

[2] An SSE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

ECL: Unusual Event

Initiating Condition: Hazardous events

Operating Mode Applicability: All

Emergency Action Levels:

Note: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

HU3.1 A tornado strike within the PROTECTED AREA.

HU3.2 Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

HU3.3 Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

HU3.4 A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

HU3.5 Lake level greater than or equal to +8.0 ft. (Plant elevation).

HU3.6 Pump bay level less than -15.0 ft.

Definitions:

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the Protected Area.

EAL HU3.2 addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

EAL HU3.3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

EAL HU3.4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

EAL HU3.5 addresses lake water level as the Turbine Building is susceptible to external flooding. +8.0 ft. corresponds to the Turbine Building floor elevation.

EAL HU3.6 addresses operability of the Emergency Diesel Generators and all Containment fan coolers. The pump bay level of -15.0 ft. represents the value at which the Emergency Diesel Generators and all Containment Fan Coolers must be declared inoperable, and is four feet above the level at which the service water pumps may begin to cavitate.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

ECL: Unusual Event

Initiating Condition: FIRE potentially degrading the level of safety of the plant.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the Unusual Event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- A Containment fire alarm is considered valid upon receipt of multiple zones (more than 1) on the FACP system (this note is applicable in Modes 1 and 2 only).

HU4.1 a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications:

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

b. The FIRE is located within **ANY** Table H-1 plant rooms or areas

HU4.2 a. Receipt of a single fire alarm with no other indications of a FIRE.

AND

b. The FIRE is located within **ANY** Table H-1 plant rooms or areas 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.

HU4.3 A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60 minutes of the initial report, alarm or indication.

HU4.4 A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Table H-1 Areas
Control Room
Containment
PAB
G05 building
13.8kV Building
Cable Spreading Room
Vital Switchgear Room
AFW Pump Room
G-01/02 Rooms
EDG Building
Service Water Pump Rooms
Façade 85’

Definitions:

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

With regard to containment fire alarms, there is constant air movement in containment due to the operation of the air handling system drawing air to the cooling units past the smoke detectors. It can reasonably be expected that a fire that burns for 15 minutes would produce sufficient products of combustion to cause fire detectors in multiple zones to alarm.

EAL HU4.1

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

EAL HU4.2

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. Except for the Containment Building in Modes 1 or 2, the 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then EAL HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

EAL HU4.3

In addition to a FIRE addressed by EAL HU4.1 or EAL HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of the ISFSI.

EAL HU4.4

If a FIRE within the plant or ISFSI PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Basis-Related Requirements from Appendix R and NFPA-805

Criterion 3 of Appendix A to 10 CFR 50 states in part that "structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

The Nuclear Safety Goal ("NSG") in NFPA 805, Section 1.3.1 states, "The nuclear safety goal is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance because a safe shutdown success path, free of fire damage, must be available to meet the nuclear safety goals, objectives and performance criteria for a fire under any plant operational mode or configuration.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). Even though PBNP has adopted the alternate approach provided by NFPA-805 in lieu of the deterministic requirements of Appendix R, the 30-minutes to verify a single alarm as used in EAL HU4.2 is considered a reasonable amount of time to determine if an actual FIRE exists without presenting a challenge to the nuclear safety performance criteria.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

ECL: Unusual Event

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE.

Operating Mode Applicability: All

Emergency Action Levels:

HU7.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a UE.

ECL: Alert

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.

Operating Mode Applicability: All

Emergency Action Levels:

HA1.1 A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.

HA1.2 A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Definitions:

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION: An act toward PBNP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

OWNER CONTROLLED AREA: The site property owned by or otherwise under the control of the licensee.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between Security Shift Supervisor and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering).

The Alert declaration will also heighten the awareness of Offsite Response Organizations, allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72.

EAL HA1.1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA.

EAL HA1.2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and offsite response organizations are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with SY-AA-102-1014, Threat Assessment and Reporting.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.

Escalation of the emergency classification level would be via IC HS1.

HA5

ECL: Alert

Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown, or shutdown.

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.

HA5.1 a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas:

Area	Mode
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B32 MCC Area	4

AND

b. Entry into the room or area is prohibited or impeded.

Definitions:

None

Basis:

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply.

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness, or even death.

The list of plant rooms or areas in EAL HA5.1 was generated from a step-by-step review of OP-3A, 3B, 3C, 5D, and 7A.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

ECL: Alert

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.

Operating Mode Applicability: All

Emergency Action Levels:

HA6.1 An event has resulted in plant control being transferred from the Control Room to AOP local control stations.

Definitions:

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS6.

ECL: Alert

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.

Operating Mode Applicability: All

Emergency Action Levels:

HA7.1 Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Definitions:

HOSTILE ACTION: An act toward PBNP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for an Alert.

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Levels:

HS1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

Definitions:

HOSTILE ACTION: An act toward PBNP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervisor and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize offsite response organization resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.

Escalation of the emergency classification level would be via IC HG1.

ECL: Site Area Emergency

Initiating Condition: Inability to control a key safety function from outside the Control Room.

Operating Mode Applicability: All

Emergency Action Levels:

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

HS6.1 a. An event has resulted in plant control being transferred from the Control Room to AOP local control stations.

AND

b. Control of **ANY** of the following key safety functions is not reestablished within 15 minutes.

- Reactivity control
- Core cooling
- RCS heat removal

Definitions:

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within 15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

Escalation of the emergency classification level would via IC FG1 or CG1.

ECL: Site Area Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.

Operating Mode Applicability: All

Emergency Action Levels:

HS7.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Definitions:

HOSTILE ACTION: An act toward PBNP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a Site Area Emergency.

ECL: General Emergency

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility.

Operating Mode Applicability: All

Emergency Action Levels:

HG1.1 a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

AND

b. **EITHER** of the following has occurred:

1. **ANY** of the following safety functions cannot be controlled or maintained.

   - Reactivity control
   - Core cooling
   - RCS heat removal

   OR

2. Damage to spent fuel has occurred or is IMMINENT.

Definitions:

HOSTILE ACTION: An act toward PBNP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between Security Shift Supervisor and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.

ECL: General Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.

Operating Mode Applicability: All

Emergency Action Levels:

HG7.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Definitions:

HOSTILE ACTION: An act toward PBNP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a General Emergency.

11 SYSTEM MALFUNCTION ICS/EALS

ECL: Unusual Event

Initiating Condition: Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SU1.1 Loss of **ALL** offsite AC power capability to 1(2)A-05 and 1(2)A-06 for 15 minutes or longer.

Definitions:

None

Basis:

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

Note: with respect to this EAL, "Station Blackout is Unit 1(2) specific."

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation of the emergency classification level would be via IC SA1.

ECL: Unusual Event

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SU3.1 An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.

- Reactor Power
- RCS / Pressurizer Level
- RCS / Pressurizer Pressure
- Core Exit / RCS Temperature
- Level in at least one steam generator
- Steam Generator Auxiliary Feed Water Flow

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via IC SA3.

ECL: Unusual Event

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

SU4.1 Failed Fuel Monitor 1(2)RE-109 reading greater than 750 mR/hr.

SU4.2 Sample analysis indicates that a RCS Specific Activity value is greater than an allowable limit specified in Technical Specifications as indicated by **ANY** of the following conditions:

a. Dose Equivalent I-131 greater than 50 μCi/gm

OR

b. Dose Equivalent I-131 greater than 0.5 μCi/gm but less than or equal to 50 μCi/gm for greater than 48 hours

OR

c. Dose Equivalent Xe-133 greater than 300 μCi/gm for greater than 48 hours

Definitions:

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specification 3.4.16 for longer than the allowed completion time. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

ECL: Unusual Event

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SU5.1 RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.

SU5.2 RCS identified leakage greater than 25 gpm for 15 minutes or longer.

SU5.3 Leakage from the RCS to a location outside containment, or Steam Generator tube leakage, greater than 25 gpm for 15 minutes or longer.

Definitions:

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

EAL SU5.1 and EAL SU5.2 are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications).

EAL SU5.3 addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These EALs thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

The leak rate values for each EAL were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). EAL SU5.1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. For PBNP, an emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

ECL: Unusual Event

Initiating Condition: Automatic or manual trip fails to shutdown the reactor.

Operating Mode Applicability: 1

Emergency Action Levels:

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

SU6.1 a. An automatic trip did not shutdown the reactor.

AND

b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.

SU6.2 a. A manual trip 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 is successful in shutting down the reactor.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles 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.

NOTE: For PBNP, the phrase "at the reactor control consoles" means the reactor trip pushbuttons on the following panels:

- Unit 1 on panels 1C04 and C01
- Unit 2 on panels 2C04 and C02

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions in the Control Room to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch). 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 in the Control Room is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

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 reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.

- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

ECL: Unusual Event

Initiating Condition: Loss of all onsite or offsite communications capabilities.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

SU7.1 Loss of **ALL** of the following onsite communication methods:
- Plant Public Address System (Gai-Tronics)
- Commercial Phones
- PBX Phones
- Security Radio
- Portable Radios

SU7.2 Loss of **ALL** of the following offsite response organization communications methods:
- Nuclear Accident Reporting System (NARS)
- Commercial Phones
- PBX Phones
- Satellite Phones
- Manitowoc County Sheriff's Department Radio

SU7.3 Loss of **ALL** of the following NRC communications methods:
- FTS Phone System
- Commercial Phones
- PBX Phones
- Satellite Phones

Definitions:

None

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to offsite response organizations and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL SU7.1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL SU7.2 addresses a total loss of the communications methods used to notify all offsite response organizations of an emergency declaration. The offsite response organizations referred to here are the State of Wisconsin, Manitowoc County, and Kewaunee County.

EAL SU7.3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

ECL: Unusual Event

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

SU8.1 a. Failure of containment to isolate when required by an actuation signal.

AND

b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal.

SU8.2 a. Containment pressure greater than 25 psig.

AND

b. Less than one full train of Containment Cooling System equipment is operating per design for 15 minutes or longer.

Definitions:

None

Basis:

This IC addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For EAL SU8.1, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

EAL SU8.2 addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. During a design basis accident, a minimum of two containment accident fan cooler units with their accident fans running and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits. Each containment spray train is a containment spray pump, spray header, nozzles, valves and piping. Each containment accident fan cooler unit consists of cooling coils, accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path.

The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

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:

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SA1.1 a. AC power capability to 1(2)A-05 **AND** 1(2)A-06 is reduced to a single power source for 15 minutes or longer.
AND

b. Any additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

- Normal Unit 1(2) offsite power sources include:
  - 345 KVAC 1(2)X-03 through the 13.8 KVAC system to the LVSAT 1(2)X-04
  - 345 KVAC backfed through the 19 KVAC system to the UAT 1(2)X-02

- Normal Unit 1(2) onsite power sources consist of:
  - emergency diesel generators
  - gas turbine generator
  - unit main turbine generator
  - power supplied from the opposite unit

An “AC power source” is a source recognized in AOPs and EOPs (including Beyond Design Basis event procedures), and capable of supplying required power to an emergency bus. Some examples of this Initiating Condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

ECL: Alert

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SA3.1 a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.

- Reactor Power
- RCS / Pressurizer Level
- RCS / Pressurizer Pressure
- Core Exit / RCS Temperature
- Levels in at least one steam generator
- Steam Generator Auxiliary 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

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via ICs FS1 or IC RS1.

ECL: Alert

Initiating Condition: Automatic or manual trip fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down 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:

SA6.1 a. An automatic or manual trip did not shutdown the reactor.

AND

b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

NOTE: For PBNP, the phrase "at the reactor control consoles" means the reactor trip pushbuttons on the following panels:

- Unit 1 on panels 1C04 and C01
- Unit 2 on panels 2C04 and C02

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (e.g., locally opening breakers). Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles."

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 SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

ECL: Alert

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

SA9.1 a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Lake level greater than or equal to +9.0 ft. (Plant elevation)
- Pump bay level less than -19.0 ft.
- Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

AND

b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or

- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Definitions:

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria SA9.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

High lake water level conditions that may have resulted in a plant VITAL AREA being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant SAFETY SYSTEMS. Lake water level at +9.0 feet corresponds to the license basis flood elevation and is one foot above the Turbine Building floor elevation. The low pump bay level threshold corresponds to the level that is calculated to correspond to the onset of cavitation of the service water pumps.

Escalation of the emergency classification level would be via IC FS1 or RS1.

ECL: Site Area Emergency

Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SS1.1 Loss of **ALL** offsite and **ALL** onsite AC power to 1(2)A-05 and 1(2)A-06 for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

For the purpose of classification under this EAL, evaluation of power sources should be made on each unit individually.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to safety-related 4160 VAC busses. Even though a safety-related 4160 VAC bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

ECL: Site Area Emergency

Initiating Condition: Loss of all Vital DC power for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SS2.1 Indicated voltage is less than 115 VDC on **ALL** Vital DC busses D-01, D-02, D-03, and D-04 for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG2.

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:

SS6.1 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:

- Conditions requiring entry into Core Cooling – Red Path (CSP-C.1) are met.
- Conditions requiring entry into Heat Sink – Red Path (CSP-H.1) are met.

Definitions:

None

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: General Emergency

Initiating Condition: Prolonged loss of all offsite and all onsite AC power to emergency buses.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the General Emergency promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.

SG1.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to 1(2)A-05 and 1(2)A-06.

AND

b. **EITHER** of the following:

- Restoration of at least one AC emergency bus in less than 4 hours is not likely.
- Conditions requiring entry into Core Cooling – Red Path (CSP-C.1) are met.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

For the purpose of classification under this EAL, evaluation of power sources should be made on each unit individually.

This IC addresses a prolonged loss of all power sources to AC emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

ECL: General Emergency

Initiating Condition: Loss of all AC and Vital DC power sources for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the General Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SG2.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to 1(2)A-05 and 1(2)A-06 for 15 minutes or longer.

AND

b. Indicated voltage is less than 115 VDC on **ALL** Vital DC busses D-01, D-02, D-03 and D-04 for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a concurrent and prolonged loss of both AC and Vital DC power. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of Vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both AC and DC power will lead to multiple challenges to fission product barriers.

If mitigative strategies establish emergency power to any bus listed in the EAL, the EAL threshold for this Initiating Condition is not met.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC .... Alternating Current
AOP .... Abnormal Operating Procedure
ATWS .... Anticipated Transient Without Scram
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CNMT .... Containment
CSF .... Critical Safety Function
CSFST .... Critical Safety Function Status Tree
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
FEMA .... Federal Emergency Management Agency
GE .... General Emergency
IC .... Initiating Condition
ID .... Inside Diameter
ISFSI .... Independent Spent Fuel Storage Installation
Keff .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LOCA .... Loss of Coolant Accident
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW .... Megawatt
NEI .... Nuclear Energy Institute
NRC .... Nuclear Regulatory Commission
NORAD .... North American Aerospace Defense Command
NUMARC[1] .... Nuclear Management and Resources Council
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Offsite Dose Calculation Manual
PA .... Protected Area
PAG .... Protective Action Guideline
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RVLIS .... Reactor Vessel Level Instrumentation System

[1] NUMARC was a predecessor organization of the Nuclear Energy Institute (NEI).

SCBA .... Self-Contained Breathing Apparatus
SG .... Steam Generator
SI .... Safety Injection
SPDS .... Safety Parameter Display System
TEDE .... Total Effective Dose Equivalent
TSC .... Technical Support Center
UE .... Unusual Event
UFSAR .... Updated Final Safety Analysis Report
WOG .... Westinghouse Owners Group

APPENDIX B – DEFINITIONS

The following definitions are taken from Title 10, Code of Federal Regulations, and related regulatory guidance documents.

Alert: Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.

General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Unusual Event (UE): Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.

The following are key terms necessary for overall understanding the NEI 99-01 emergency classification scheme.

Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

CONTAINMENT CLOSURE: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION: An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

OWNER CONTROLLED AREA: This term is typically taken to mean the site property owned by or otherwise under the control of the licensee.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool and fuel transfer canal.

RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

ATTACHMENT 3

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 286, ADOPTION OF EMERGENCY ACTION LEVEL SCHEME PURSUANT TO NEI 99-01 REVISION 6,
"DEVELOPMENT OF EMERGENCY ACTION LEVELS FOR NON-PASSIVE REACTORS"

UPDATED DEVIATIONS AND DIFFERENCES MATRIX

99 pages follow

PBNP DEVIATIONS AND DIFFERENCES MATRIX

TABLE OF CONTENTS

GENERAL COMMENTS

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
GLOBAL #1	References to NEI 99-01	Replaced with PBNP	Difference	Convert generic guidance to PBNP specific.	None
GLOBAL #2	Effective date	Replaced with **TBD, 2018**	Difference	Convert generic guidance to PBNP specific.	None
GLOBAL #3	Defined terms in Appendix B; Title Case	Defined terms in Appendix B; Upper Case	Difference	All defined terms in Appendix B used in the document are in upper case (CAPs) to indicate that the terms are defined.	None
GLOBAL #4	BWR specific references	BWR references removed	Difference	PBNP is a PWR	None
GLOBAL #5	Recognition Category A-Abnormal Radiation Levels/Radiological Effluent category and Emergency Action Levels; AU, AA, AS, and AG	Recognition Category R-Abnormal Radiation Levels/Radiological Effluent category and Emergency Action Levels; RU, RA, RS, and RG	Difference	PBNP implemented the optional designation of "R" for radiological related items to maintain continuity with previous practice at PBNP.	None
GLOBAL #6	Permanently Defueled Section	Deleted references to Permanently Defueled Station	Difference	Not Applicable to PBNP	None
GLOBAL #7	Acknowledgments, Notice and Executive Summary	Deleted	Difference	Not Applicable to PBNP	None
GLOBAL #8	Parameters or indications listed in EALs	Some parameters or indications listed in EALs were placed in tables or bulletized lists.	Difference	Tables or bullets were created to present PBNP-specific information in a manner familiar to and desired by scheme users.	None
GLOBAL #9	Site specific information or indication statements	Site specific information or indication statements were replaced with PBNP information or indications where applicable and the statement deleted.	Difference	Compliance with intent of the guidance.	None
GLOBAL #10	Operating Mode Applicability lists mode names (i.e., Power Operation, Startup)	Operating Mode Applicability lists mode numbers (i.e., Modes 1 and 2)	Difference	Mode numbers used for consistency with PBNP procedures and training.	None
GLOBAL #11	Developer's Notes	Developer's Notes deleted	Difference	Developer's notes are not reflected in the implementation of the EALs.	None
GLOBAL #12	Example EAL statement	"Example" deleted from statement	Difference	In adopting the EAL, the "example" status is no longer applicable.	None
GLOBAL #13	The following terms: "all, any" are sometimes capitalized and/or bolded in ICs and EALs	Consistently capitalized and bolded the following terms: "all, any" in ICs and EALs.	Difference	Capitalized and bolded conditional terms in ICs and EALs for consistency based on user feedback.	None
GLOBAL #14	Defined terms are only listed in APPENDIX B - DEFINITIONS	Defined terms are also listed as in separate section of each IC/EAL where the defined terms are used.	Difference	Aid to the user to present all needed information within the same section of the Basis document.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
COVER PAGE	Development of Emergency Action Levels for Non-Passive Reactors	Point Beach Emergency Action Level Bases Document	Difference	Changes made to adapt the generic NEI guidance to a PBNP-specific document	None
Introduction	Acknowledgments, Notice and Executive Summary	Deleted	Difference	Not Applicable to PBNP	None
TOC	1. Regulatory Background	1. Development of Emergency Action Levels	Difference	Title change	None
TOC	1.1 Operating Reactors	1.1 Regulatory Background	Difference	Title change	None
TOC	1.2 Permanently Defueled Station	Deleted section	Difference	Not Applicable to PBNP	None
TOC	1.3 Independent Spent Fuel Storage Installation (ISFSI)	1.2 Independent Spent Fuel Storage Installation (ISFSI)	Difference	Re-numbered	None
TOC	1.4 NRC Order EA-12-051	1.3 NRC Order EA-12-051	Difference	Re-numbered	None
TOC	1.5 Applicability of Advance and Small Modular Reactor Designs	Deleted section	Difference	Not Applicable to PBNP	None
TOC	3.Design of the NEI 99-01 Emergency Classification Scheme	3. Design of the PBNP Emergency Classification Scheme	Difference	Title Change	None
TOC	3.3 NSSS Design Differences	Deleted section	Difference	Changes made to adapt the generic NEI guidance to a PBNP-specific document	None
TOC	3.4 Organization and Presentation of Generic Information	Changed to 3.3 PBNP 3.4 Organization and Presentation of Generic Information	Difference	Changes made to adapt the generic NEI guidance to a PBNP-specific document	None
TOC	4.0 Site-Specific Scheme Development	4.0 PBNP Scheme Development	Difference	Title change	None
TOC	4.4; 4.5; 4.6; 4.8	Deleted sections	Difference	Changes made to adapt the generic NEI guidance to a PBNP-specific document	None
TOC	4.7 Developer and User Feedback				None
TOC	Appendix C-Permanently Defueled Station ICs/EALs	Deleted section	Difference	Changes made to adapt the generic NEI guidance to a PBNP-specific document	None
1.1	Regulatory Background	Regulatory Background	Difference	Changes made to adapt the generic NEI guidance to a PBNP-specific document and removed developer information	None
1.2	Permanently Defueled Station	Section deleted	Difference	Not Applicable to PBNP	None
1.3	1.3 Independent Spent Fuel Storage Installation (ISFSI)	1.2 Independent Spent Fuel Storage Installation (ISFSI)	Difference	Re-numbered section and renamed E-HU1 to EU1 to prevent user confusion.	None
1.4	1.4 NRC Order EA-12-051	1.3 NRC Order EA-12-051	Difference	Re-numbered and removed wording to add these readings once the instruments are installed (PBNP installation completed).	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
1.5	Applicability to Advanced and Small Modular Reactor Designs	Section deleted	Difference	Not Applicable to PBNP	None
2	KEY TERMINOLOGY USED IN NEI 99-01	KEY TERMINOLOGY USED IN NEI 99-01	Difference	Minor changes to reflect PBNP-specific implementation.	None
3	DESIGN OF THE NEI 99-01 EMERGENCY CLASSIFICATION SCHEME	DESIGN OF THE PBNP EMERGENCY CLASSIFICATION SCHEME	Difference	Changes made to adapt the generic NEI guidance to a PBNP-specific document	None
3.1	Assignment of Emergency Classification Levels (ECLs)	Assignment of Emergency Classification Levels (ECLs)	Difference	Changes made to adapt the generic NEI guidance to a PBNP-specific document, removed references to BWRs, and removed developer information.	None
3.2	Types of Initiating Conditions and Emergency Action Levels	Types of Initiating Conditions and Emergency Action Levels	Verbatim		None
3.3	Text referring to NSSS design differences for various types or nuclear plants; Developer guidance	Deleted	Difference	Guidance is now PBNP specific	None
3.4	Organization and Presentation of Generic Information	PBNP-Specific Organization and Presentation of Generic Information	Difference	Renumbered to 3.3, made PBNP-specific, and deleted developer information	None
3.5	Mode of Applicability Matrix; Typical PWR Operating Modes	Deleted "Permanently Defueled" section of matrix; replaced Typical PWR Operating Modes with PBNP Operating Modes	Difference	Renumbered to 3.4, removed BWR information, removed permanently defueled, and inserted PBNP Operating Modes to comply with the intent of the document. Global comment #5	V1
4	Site Specific Scheme Development Guidance	PBNP Scheme Development	Difference	Updated to reflect PBNP specific scheme development process.	None
5	GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS	GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS	Difference	Added text from Section IV.H.7 of NSIR/DPR-ISG-01 explaining how to treat concurrent time periods when making an emergency declaration. Information was added to address a frequently asked question by the PBNP operators.	V2
6 - 11	Recognition Category IC/EAL Matrixes	removed	Difference	Matrixes were intended for use by EAL developers. Not included in licensee scheme.	None

ABNORMAL RAD LEVELS / RADIOACTIVE EFFLUENT ICS/EALS

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AU1	Recognition Category: AU1	RU1	Difference	Global Comment #5	None
	Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.	Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.	Difference	Global Comment #9	None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td>AU1 (cont.)</td><td>(1) Reading on ANY effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer: (site-specific monitor list and threshold values corresponding to 2 times the controlling document limits)</td><td>(1) Reading on ANY of the following effluent radiation monitors greater than the reading shown for 60 minutes or longer:
<table>
<tr><th><u>Monitor</u></th><th><u>Reading</u></th></tr>
<tr><td>1(2)-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation</td><td>1.4E-2 µCi/cc</td></tr>
<tr><td>2-RE-305 CTMNT Purge Exhaust Low Range Gas with both purge and GS building ventilation in operation</td><td>9.4E-3 µCi/cc</td></tr>
<tr><td>2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation</td><td>9.4E-3 µCi/cc</td></tr>
<tr><td>2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only GS building ventilation in operation</td><td>2.8E-2 µCi/cc</td></tr>
<tr><td>2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only forced vent of containment</td><td>1.0E+1 µCi/cc</td></tr>
<tr><td>2-RE-309 CTMNT Purge Exhaust High Range Gas with only forced vent of containment</td><td>1.0E+1 µCi/cc</td></tr>
<tr><td>RE-315 AB Exhaust Low Range Gas</td><td>5.4E-3 µCi/cc</td></tr>
<tr><td>RE-317 AB Exhaust Mid-Range Gas</td><td>5.4E-3 µCi/cc</td></tr>
<tr><td>RE-325 Drumming Area Exhaust Low Range Gas</td><td>8.4E-3 µCi/cc</td></tr>
<tr><td>RE-327 Drumming Area Exhaust Mid-Range Gas</td><td>8.4E-3 µCi/cc</td></tr>
<tr><td>1(2) RE-229 Service Water Overboard</td><td>2.3E-3 µCi/cc</td></tr>
</table></td><td>Difference</td><td>See Global Comments #8, 9, 12, & 13.

Reworded EAL statement to remove operator confusion as to whether they needed to multiply the values of the following table by 2 or if the value provided already was 2X. Wording now matches wording of RS1 and RG1 allowing for easier operator progression through the EALs.</td><td>V3 & V4</td></tr>
</table>

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AU1 (cont.)	(2) Reading on ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	(2) Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	Difference	Global Comment #13	None
	(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.	(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODCM limits for 60 minutes or longer.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AU2	Recognition Category: AU2	RU2	Difference	Global Comment #5 & 14	None
	Initiating Condition: UNPLANNED loss of water level above irradiated fuel.	UNPLANNED loss of water level above irradiated fuel.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None
	(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following: (site-specific level indications). **AND**	(1) a UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: • Spent fuel pool low water level alarm • Visual observation **AND**	Difference	Global Comment #9, 12 & 13	None
	b. UNPLANNED increase in area radiation levels as indicated by ANY of the following radiation monitors. (site-specific list of area radiation monitors)	b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors. • RE-105 SFP Area Low Range Radiation Monitor • RE-135 SFP Area High Range Radiation Monitor • 1(2) RE-102 El. 66' CONTAINMENT Low Range Monitor	Difference	Global Comments #9 & 13 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AA1	Recognition Category: AA1	RA1	Difference	Global Comment #5 & 14	None
	Initiating condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.	Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AA1 (cont.)	(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	(1) Reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer: Monitor — Reading 1(2)-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation — 6.0E+0 µCi/cc 1(2)-RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation — 6.0E+0 µCi/cc 2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation — 4.0E+0 µCi/cc 2-RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation — 4.0E+0 µCi/cc 2-RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation — 1.2E+1 µCi/cc RE-317 AB Exhaust Mid-Range Gas — 1.0E+0 µCi/cc RE-319 AB Exhaust High Range Gas — 1.0E+0 µCi/cc RE-327 Drumming Area Exhaust Mid-Range Gas — 1.6E+0 µCi/cc	Difference	Global Comment #8, 9, 12 & 13	V5

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AA1 (cont.)	(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY.	Difference	Global Comment #9	None
	(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.	(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.	Difference	Global Comment #9	
	(4) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	(4) Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AA2	Recognition Category: AA2	RA2	Difference	Global Comment #5 & 14	None
	Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.	Significant lowering of water level above, or damage to, irradiated fuel.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None
	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.	Verbatim		None
	(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following radiation monitors: (site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)	(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by reading on **ANY** of the following radiation monitors greater than the reading shown: Monitor / Reading: RE-105 SFP Area Low Range Radiation Monitor — 4 R/hr 1(2)-RE-126 Containment High Radiation Monitor — 7 R/hr 1(2)-RE-127 Containment High Radiation Monitor — 7 R/hr 1(2)-RE-128 Containment High Radiation Monitor — 7 R/hr	Difference	Global Comment #8, 9, 12 & 13	V6
	(3) Lowering of spent fuel pool level to (site-specific Level 2 value). [See Developer Notes	(3) Lowering of spent fuel pool level to 49 ft. 0 in.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V7

PBNP DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="5">AA3</td><td>Recognition Category: AA3</td><td>RA3</td><td>Difference</td><td>Global Comment #5 & 14</td><td>None</td></tr>
<tr><td>Initiating Condition: Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.</td><td>Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: All</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>(1) Dose rate greater than 15 mR/hr in ANY of the following areas:
• Control Room
• Central Alarm Station
• (other site-specific areas/rooms)</td><td>(1) Dose rate greater than 15 mR/hr in ANY of the following areas:
• Control Room (RE-101)
• Central Alarm Station AND Secondary Alarm Station (by survey)</td><td>Difference</td><td>Global Comment #9, 12 & 13</td><td>None</td></tr>
<tr><td>(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>(2) An UNPLANNED event results in radiation levels that prohibit or impede access to ANY of the following plant rooms or areas:

<table><tr><th colspan="2">Table R-2
SAFE OPS, S/D, C/D AREAS</th></tr><tr><th>Area/Building</th><th>MODE</th></tr><tr><td>U1 VCT Area</td><td>3 / 4 / 5</td></tr><tr><td>U2 VCT Area</td><td>3 /4 / 5</td></tr><tr><td>U1 Primary Sample area</td><td>3</td></tr><tr><td>U2 Primary Sample area</td><td>3</td></tr><tr><td>CCW HX Room</td><td>4 / 5</td></tr><tr><td>C-59 area</td><td>3 / 4 / 5</td></tr><tr><td>Pipeway 2, 8 ft. Elev.</td><td>3 / 4</td></tr><tr><td>Pipeway 3, 8 ft. Elev.</td><td>3 / 4</td></tr><tr><td>1/2B32 MCC Area</td><td>4</td></tr></table></td><td>Difference</td><td>Global Comment #8, 9 & 13

Intent and meaning of the EALs are not altered.</td><td>V8</td></tr>
</table>

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AS1	Recognition Category: AS1	RS1	Difference	Global Comment #5 & 14	None
	Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AS1 (cont.)	(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	(1) Reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer: Monitor — Reading 1(2)-RE-307 CTMNT Purge Exhaust Mid-Range Gas with only containment purge in operation — 6.0E+1 µCi/cc 1(2)-RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation — 6.0E+1 µCi/cc 2-RE-307 CTMNT Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation — 4.0E+1 µCi/cc 2-RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation — 4.0E+1 µCi/cc 2-RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation — 1.2E+2 µCi/cc RE-317 AB Exhaust Mid-Range Gas — 1.0E+1 µCi/cc RE-319 AB Exhaust High Range Gas — 1.0E+1 µCi/cc RE-327 Drumming Area Exhaust Mid-Range Gas — 1.6E+1 µCi/cc	Difference	Global Comment #8, 9, 12 & 13	V5

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AS1 (cont.)	(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY.	Difference	Global Comment #3 & 9	None
	(3) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.	(3) Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.	Difference	Global Comment #3 & 9 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AS2	Recognition Category: AS2	RS2	Difference	Global Comment #5	None
	Initiating Condition: Spent fuel pool level at (site-specific Level 3 description).	Spent fuel pool level at 40 ft. 8 in.	Difference	Global Comment #9	V7
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Lowering of spent fuel pool level to (site-specific Level 3 value).	(1) Lowering of spent fuel pool level to 40 ft. 8 in.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V7

PBNP DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="4">AG1</td><td>Recognition Category: AG1</td><td>RG1</td><td>Difference</td><td>Global Comment #5 & 14</td><td>None</td></tr>
<tr><td>Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.</td><td>Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: All</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:
(site-specific monitor list and threshold values)</td><td>(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:
<table>
<tr><th>Monitor</th><th>Reading</th></tr>
<tr><td>1(2)-RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation</td><td>6.0E+2 µCi/cc</td></tr>
<tr><td>2-RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation</td><td>4.0E+2 µCi/cc</td></tr>
<tr><td>2-RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation</td><td>1.2E+3 µCi/cc</td></tr>
<tr><td>RE-317 AB Exhaust Mid-Range Gas</td><td>1.0E+2 µCi/cc</td></tr>
<tr><td>RE-319 AB Exhaust High Range Gas</td><td>1.0E+2 µCi/cc</td></tr>
</table></td><td>Difference</td><td>Global Comment #8, 9, 12 & 13</td><td>V5</td></tr>
</table>

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AG1 (cont.)	(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY.	Difference	Global Comment #3 & 9	None
	(3) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.	(3) Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.	Difference	Global Comment #3 & 9 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
AG2	Recognition Category: AG2	RG2	Difference	Global Comment #5	None
	Initiating Condition: Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer.	Spent fuel pool level cannot be restored to at least 40 ft. 8 in. for 60 minutes or longer.	Difference	Global Comment #9	V7
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer.	(1) Spent fuel pool level cannot be restored to at least 40 ft. 8 in. for 60 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V7

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CU1	Recognition Category: CU1	CU1	Verbatim	Global Comment #14	None
	Initiating Condition: UNPLANNED loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory for 15 minutes or longer.	UNPLANNED loss of reactor vessel/RCS inventory for 15 minutes or longer	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	Global Comment #10	None
	(1) UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [PWR] or RPV [BWR]) level less than a required lower limit for 15 minutes or longer.	(1) UNPLANNED loss of reactor coolant results in reactor vessel/RCS level less than a required lower limit for 15 minutes or longer	Difference	Global Comment #4 & 12	None
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored. AND	(2) a. Reactor vessel/RCS level cannot be monitored. **AND**	Difference	Global Comment #4	None
	b. UNPLANNED increase in (site-specific sump and/or tank) levels.	b. UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V9

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CU2	Recognition Category: CU2	CU2	Verbatim	Global Comment #14	None
	Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 5, 6, Defueled	Difference	Global Comment #10	None
	(1) a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.	(1) a. AC power capability to 1(2)-A05 and 1(2)- A06 is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V10

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CU3	Recognition Category: CU3	CU3	Verbatim	Global Comment #14	None
	Initiating Condition: UNPLANNED increase in RCS temperature.	UNPLANNED increase in RCS temperature.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	Global Comment #10	None
	(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit).	(1) UNPLANNED increase in RCS temperature to greater than 200°F	Difference	Global Comment #9 & 12	V1
	(2) Loss of ALL RCS temperature and (reactor vessel/RCS [PWR] or RPV [BWR]) level indication for 15 minutes or longer.	(2) Loss of **ALL** RCS temperature and reactor vessel/RCS level indication for 15 minutes or longer	Difference	Global Comment #4 & 13 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CU4	Recognition Category: CU4	CU4	Verbatim	Global Comment #14	None
	Initiating Condition: Loss of Vital DC power for 15 minutes or longer.	Loss of Vital DC power for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	Global Comment #10	None
	(1) Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer.	(1) Indicated voltage is less than 115VDC on required Vital DC buses D-01, D-02, D-03 or D-04 for 15 minutes or longer	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V11

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CU5	Recognition Category: CU5	CU5	Verbatim		None
	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Loss of all onsite or offsite communications capabilities.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 5, 6, Defueled	Difference	Global Comment #10	None
	(1) Loss of ALL of the following onsite communication methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following onsite communication methods: • Plant Public Address System (Gai-Tronics) • Commercial Phones • PBX Phones • Security Radio • Portable Radio	Difference	Global Comment #9, 12 & 13	None
	(2) Loss of ALL of the following ORO communications methods: (site-specific list of communications methods)	(2) Loss of **ALL** of the following offsite response organization communications methods: • Nuclear Accident Reporting System (NARS) • Commercial Phones • PBX Phones • Satellite Phones • Manitowoc County Sheriff's Department Radio	Difference	Global Comment #9, 12 & 13	None
	(3) Loss of ALL of the following NRC communications methods: (site-specific list of communications methods)	(3) Loss of **ALL** of the following NRC communications methods: • FTS Phone System • Commercial Phones • PBX Phones • Satellite Phones	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CA1	Recognition Category: CA1	CA1	Verbatim	Global Comment #14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory.	Loss of reactor vessel/RCS inventory.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	Global Comment #10	None
	(1) Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory as indicated by level less than (site-specific level).	(1) Loss of reactor vessel/RCS inventory as indicated by level less than 16% on LI-447/ LI-447A	Difference	Global Comment #4, 9 & 12	V12
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 15 minutes or longer	(2) a. Reactor vessel/RCS level cannot be monitored for 15 minutes or longer	Difference	Global Comment #4	None
	AND b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory.	**AND** b. UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels due to a loss of reactor vessel/RCS inventory.	Difference	Global Comment #4, 9 & 13 Intent and meaning of the EALs are not altered.	V9

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CA2	Recognition Category: CA2	CA2	Verbatim	Global Comment #14	None
	Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 5, 6, Defueled	Difference	Global Comment #10	None
	(1) Loss of ALL offsite and ALL onsite AC Power to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite and **ALL** onsite AC Power to 1(2)-A05 and 1(2)-A06 for 15 minutes or longer.	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	V10

PBNP DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="6">CA3</td><td>Recognition Category: CA3</td><td>CA3</td><td>Verbatim</td><td>Global Comment #14</td><td>None</td></tr>
<tr><td>Initiating Condition: Inability to maintain the plant in cold shutdown.</td><td>Inability to maintain the plant in cold shutdown.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: Cold Shutdown, Refueling</td><td>Operating Mode Applicability: 5, 6</td><td>Difference</td><td>Global Comment #10</td><td>None</td></tr>
<tr><td>(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in the following table.</td><td>(1) UNPLANNED increase in RCS temperature to greater than 200°F for greater than the duration specified in the following table:</td><td>Difference</td><td>Global Comment #9 & 12</td><td>V1</td></tr>
<tr><td>
<table>
<tr><th colspan="2">Table: RCS Heat-up Duration Thres</th></tr>
<tr><th>RCS Status</th><th>Containment Closure Status</th></tr>
<tr><td>Intact (but not at reduced inventory [PWR])</td><td>Not applicable</td></tr>
<tr><td rowspan="2">Not intact (or at reduced inventory [PWR])</td><td>Established</td></tr>
<tr><td>Not Established</td></tr>
<tr><td colspan="2">* If an RCS heat removal system is in operatio frame and RCS temperature is being reduce applicable.</td></tr>
</table>
</td><td>
<table>
<tr><th colspan="2">RCS Heat-up Duration Thresho</th></tr>
<tr><th>RCS Status</th><th>Containment Closure Status</th></tr>
<tr><td>Intact (but not at reduced inventory)</td><td>Not Applicable</td></tr>
<tr><td rowspan="2">Not intact (or at reduced inventory)</td><td>Established</td></tr>
<tr><td>Not Established</td></tr>
<tr><td colspan="2">* If RHR is in operation within this time frame temperature is being reduced, the EAL is not a</td></tr>
</table>
</td><td>Difference</td><td>Global Comment #9

Replaced "RCS heat removal system" with "RHR" to reflect site-specific nomenclature familiar to the operators.</td><td>None</td></tr>
<tr><td>(2) UNPLANNED RCS pressure increase greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [PWR])</td><td>(2) UNPLANNED RCS pressure increase greater than 25 psig. (This EAL does not apply during water-solid plant conditions.)</td><td>Difference</td><td>Global Comment #4 & 9

Intent and meaning of the EALs are not altered.</td><td>V13</td></tr>
</table>

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CA6	Recognition Category: CA6	CA6	Verbatim	Global Comment #14	None
	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	Global Comment #10	None
	(1) a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager	(1) a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Lake level greater than or equal to 9.0 ft. (Plant elevation) • Other events with similar hazard characteristics as determined by the Shift Manager	Difference	Global Comment #9, 12 & 13	V14
	AND b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.	**AND** b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode. **AND**	Deviation	Adopted the revised EAL wording provided in proposed EAL FAQ 2016-02.	V15

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
	OR 1. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	2. **EITHER** of the following: • Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or • The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.	Deviation	Adopted the revised EAL wording provided in proposed EAL FAQ 2016-02. Intent and meaning of the EALs are not altered.	V15

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CS1	Recognition Category: CS1	CS1	Verbatim	Global Comment #14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory affecting core decay heat removal capability.	Loss of reactor vessel/RCS inventory affecting core decay heat removal capability.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	Global Comment #10	None
	(1) a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level).	Not used	Difference	Global Comment #9 & 12 PBNP design and operation of water level instrumentation is such that this level value cannot be accurately determined during Cold Shutdown or Refueling modes, therefore this EAL was not included.	None
	(1) a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level).	Not used	Difference	Global Comment #4 & 9 PBNP design and operation of water level instrumentation is such that this level value cannot be accurately determined during Cold Shutdown or Refueling modes, therefore this EAL was not included.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CS1 (cont.)	(3) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer. **AND**	(1) a. Reactor vessel/RCS level cannot be monitored for 30 minutes or longer. **AND**	Difference	Global Comment #4	None
	b. Core uncovery is indicated by **ANY** of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [PWR] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications)	b. Core uncovery is indicated by **ANY** of the following: • Containment High Radiation Monitor (1(2)-RE-126, RE-127, or RE-128) reading greater than 100 R/hr • Erratic source range monitor indication • UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery	Difference	Global Comment #9 &13 Intent and meaning of the EALs are not altered.	V9 & V16

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
CG1	Recognition Category: CG1	CG1	Verbatim	Global Comment #14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory affecting fuel clad integrity with containment challenged.	Loss of reactor vessel/RCS inventory affecting fuel clad integrity with containment challenged.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	Global Comment #10	None
	(1) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level) for 30 minutes or longer. **AND** b. ANY indication from the Containment Challenge Table (see below).	b. Not used	Difference	Global Comment #4, 9, 12 & 13 PBNP design and operation of water level instrumentation is such that this level value cannot be accurately determined during Cold Shutdown or Refueling modes, therefore this EAL was not included.	None
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer.	(1) a. Reactor vessel/RCS level cannot be monitored for 30 minutes or longer.	Difference	Global Comment #4	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
	AND b. Core uncovery is indicated by ANY of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [PWR] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery	**AND** b. Core uncovery is indicated by **ANY** of the following: • Containment High Radiation Monitor (1(2)-RE-126, RE-127, or RE-128) reading greater than 100 R/hr • Erratic source range monitor indication • UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery	Difference	Global Comment #8, 9 & 13	V8 &V16
	AND c. **ANY** indication from the Containment Challenge Table (see below).	**AND** c. **ANY** indication from Containment Challenge Table C-1	Difference	Global Comment #8 & 9	None
	Containment Challenge Table ONTAINMENT CLOSURE not established* :xplosive mixture) exists inside containment NPLANNED increase in containment pressure econdary containment radiation monitor reading ;ite specific value) [BWR]	**Containment Challenge Table C** • CONTAINMENT CLOSURE not established* • 6% H2 concentration exists inside containm • UNPLANNED increase in containment press	Difference	Global Comment #8 & 9	V17
	* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.	*If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.	Verbatim	Intent and meaning of the EALs are not altered.	

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

PBNP DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="4">E-HU1</td><td>Recognition Category: E-HU1</td><td>EU1</td><td>Difference</td><td>PBNP changed Recognition Category to maintain continuity with previous practice at PBNP.
Global Comment #14</td><td>None</td></tr>
<tr><td>Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.</td><td>Damage to a loaded cask CONFINEMENT BOUNDARY.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: All</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.</td><td>(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than the values shown below on the surface of the spent fuel cask.
<table><tr><td colspan="2">32 PT DSC</td></tr><tr><td>Front Surface</td><td>1700 mrem/ hr</td></tr><tr><td>Door Centerline</td><td>400 mrem/ hr</td></tr><tr><td>End Shield Wall Exterior</td><td>12 mrem/ hr</td></tr><tr><td colspan="2">VSC-24</td></tr><tr><td>Sides</td><td>200 mrem/ hr</td></tr><tr><td>Top</td><td>400 mrem/ hr</td></tr><tr><td>Air Inlets</td><td>700 mrem/ hr</td></tr><tr><td>Air Outlets</td><td>200 mrem/ hr</td></tr></table></td><td>Difference</td><td>Global Comment #8, 9 & 12

Intent and meaning of the EALs are not altered.</td><td>V18</td></tr>
</table>

FISSION PRODUCT BARRIER ICS/EALS

The following section is configured in a manner that is different from the Fission Product Barrier Tables in the PBNP EAL Technical Bases Document. Where the Technical Bases Document evaluates all three fission product barriers simultaneously for a specific sub-category, this matrix presents each fission product barrier individually for all sub-categories. The significance of this presentation is that where the fission product barrier table in the Technical Bases Document moves vertically through the sub-categories, this matrix moves horizontally.

PBNP DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th colspan="6">Fission Product Barrier Emergency Classifications</th><th></th></tr>
<tr><th colspan="3">NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td colspan="3">Table 9-F-1: Recognition Category "F" Initiating Condition Matrix</td><td rowspan="4">Deleted</td><td rowspan="4">Difference</td><td rowspan="4">Deleted per developer note. Mode applicability carried over onto Table 9-F-1 EAL listing.
Global Comment #10</td><td rowspan="4">None</td></tr>
<tr><td>Alert</td><td>Site Area Emergency</td><td>General Emergency</td></tr>
<tr><td>Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.</td><td>Loss or Potential Loss of any two barriers.</td><td>Loss of any two barriers and Loss or Potential Loss of the third barrier.</td></tr>
<tr><td>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</td><td>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</td><td>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</td></tr>
<tr><td colspan="3">Table 9-F-2: BWR EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Deleted</td><td>Difference</td><td>Global Comment #4</td><td>None</td></tr>
<tr><td colspan="3">Table 9-F-3: PWR EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Table 9-F-1: EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Difference</td><td>Renumbered and re-labeled due to deletion of Tables 9-F-1 & 2.
Added
Global Comment #9</td><td>None</td></tr>
<tr><td colspan="3">Basis Information For PWR EAL Fission Product Barrier Table 9-F-3 Developer Notes.</td><td>Deleted Developer Notes</td><td>Difference</td><td>Transform generic NEI 99-01 guidance into PBNP specific application.</td><td>None</td></tr>
<tr><td colspan="3">Figure 9-F-4: PWR Containment Integrity or Bypass Example</td><td>Figure 9-F-2: Containment Integrity or Bypass Example</td><td>Difference</td><td>Renumbered and re-labeled due to deletion of prior Tables.
Global Comment #9</td><td>None</td></tr>
</table>

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Fuel Clad Barrier						
Table 9-F-1	**NEI 99-01 Rev. 6**		**PBNP**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
Critical Safety Function Status	Not Used	Not Used	A. Conditions requiring entry into Core Cooling RED Path (CSP C.1) are met.	A. Conditions requiring entry into Core Cooling ORANGE Path (CSP C.2) are met. **OR** B. Conditions requiring entry into Heat Sink RED Path (CSP H.1) are met.	Difference	V19 - CSF status consolidated into a separate sub-category to allow operators to quickly assess barrier status via CSF status at one time. V20 - CSF status used consistent with recommendations for Westinghouse ERG plants.
1. RCS or SG Tube Leakage	Not Applicable	A. RCS/reactor vessel level less than (site-specific level).	Not Applicable	Not Applicable	Difference	CSF status used consistent with recommendations for Westinghouse ERG plants, but listed in separate sub-category 1.
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).	Sub-category not used	Sub-category not used	Difference	CSF status consolidated into a separate sub-category to allow operators to quickly assess barrier status via CSF status at one time. CSF status used consistent with recommendations for Westinghouse ERG plants.

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Fuel Clad Barrier						
Table 9-F-1	**NEI 99-01 Rev. 6**		**PBNP**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
3. RCS Activity / Containment Radiation	A. Containment radiation monitor reading greater than (site-specific value). **OR** B. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).	Not Applicable	A. Containment radiation monitor reading greater than 577 R/hr indicated on **ANY** of the following. • 1(2)-RE-126 • 1(2)-RE-127 • 1(2)-RE-128 **OR** B. 1(2)-RE-109 greater than 4,500 mR/hr	Not Applicable	Difference	V21 - Global Comment #9 V22 - Global Comment #9
4. Containment Integrity or Bypass	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Verbatim	None
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Sub-category not used	Sub-category not used	Difference	Global Comment #9 No other site-specific thresholds were identified for PBNP.
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Verbatim	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of RCS Barrier						
Table 9-F-1	NEI 99-01 Rev. 6		PBNP		Change	Justification
Sub-Category	Loss	Potential Loss	Loss	Potential Loss		
Critical Safety Function Status	Not Used	Not Used	Not Applicable	A. Conditions requiring entry into Heat Sink RED Path (CSP H.1) are met. **OR** B. Conditions requiring entry into RCS Integrity RED Path (CSP P.1) are met.	Difference	V20 - CSF status consolidated into a separate sub-category to allow operators to quickly assess barrier status via CSF status at one time. V23 - CSF status used consistent with recommendations for Westinghouse ERG plants.
1. RCS or SG Tube Leakage	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage.	Difference	Global Comment #9 CSF status consolidated into a separate sub-category to allow operators to quickly assess barrier status via CSF status at one time. CSF status used consistent with recommendations for Westinghouse ERG plants.

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of RCS Barrier						
Table 9-F-1	**NEI 99-01 Rev. 6**		**PBNP**		**Change**	**Justification**
2. Inadequate heat Removal	Not Applicable	A. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	Sub-category not used	Sub-category not used	Difference	CSF status consolidated into a separate sub-category to allow operators to quickly assess barrier status via CSF status at one time.
3. RCS Activity/ Containment Radiation	A. Containment radiation monitor reading greater than (site-specific value).	Not Applicable	Containment radiation monitor reading greater than 11 R/hr on **ANY** of the following: • 1(2)-RE-126 • 1(2)-RE-127 • 1(2)-RE-128	Not Applicable	Difference	V21 - Global Comment #9
4. Containment Integrity or Bypass	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Verbatim	None
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Sub-category not used	Sub-category not used	Difference	Global Comment #9 No other site-specific thresholds were identified for PBNP.
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Verbatim	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**PBNP**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
Critical Safety Function Status	Not Used	Not Used	Not Applicable	A. 1. Conditions requiring entry into Core Cooling RED Path (CSP C.1) are met. **AND** 2. CSP C.1 not effective within 15 minutes.	Difference	V19 - Global comment #9 CSF status consolidated into a separate sub-category to allow operators to quickly assess barrier status via CSF status at one time. CSF status used consistent with recommendations for Westinghouse ERG plants.
1. RCS or SG Tube Leakage	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable	Verbatim	None
2. Inadequate heat Removal	Not Applicable	A 1 (Site-specific criteria for entry into core cooling restoration procedure) **AND** 2 Restoration procedure not effective within 15 minutes.	Sub-category not used	Sub-category not used	Difference	CSF status consolidated into a separate sub-category to allow operators to quickly assess barrier status via CSF status at one time.

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	NEI 99-01 Rev. 6		PBNP		Change	Justification
Sub-Category	Loss	Potential Loss	Loss	Potential Loss		
3. RCS Activity / Containment Radiation	Not Applicable	A. Containment radiation monitor reading greater than (site-specific value).	Not Applicable	A. Containment radiation monitor reading greater than 18,500 R/hr indicated on **ANY** of the following. • 1(2)-RE-126 • 1(2)-RE-127 • 1(2)-RE-128	Difference	V21 - Global Comment #9
4. Containment Integrity or Bypass	A. Containment isolation is required **AND EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment	A. Containment pressure greater than (site-specific value) **OR** B. Explosive mixture exists inside containment **OR** C 1. Containment pressure greater than (site-specific pressure setpoint) **AND** 2. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.	A. Containment isolation is required **AND EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment pressure greater than 60 psig. **OR** B. 6% H^2 inside containment. **OR** C. 1. Containment pressure greater than 25 psig. **AND** 2. Less than one full train of depressurization equipment is operating per design for 15 minutes or longer.	Difference	V24 - Global Comment #9 V17 V25

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**PBNP**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
5.Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Sub-category not used	Sub-category not used	Difference	Global Comment #9 No other site-specific thresholds were identified for PBNP.
6.Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	B. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	C. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	D. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	Verbatim	None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HU1	Recognition Category: HU1	HU1	Verbatim	Global Comment #14	None
	Initiating Condition: Confirmed SECURITY CONDITION or threat.	Confirmed SECURITY CONDITION or threat.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).	(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the Security Shift Supervisor.	Difference	Global Comment #9 & 12	None
	(2) Notification of a credible security threat directed at the site.	(2) Notification of a credible security threat directed at PBNP.	Difference	Global Comment #8 & 9	None
	(3) A validated notification from the NRC providing information of an aircraft threat.	(3) A validated notification from the NRC providing information of an aircraft threat.	Difference	Global Comment #8 & 9 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HU2	Recognition Category: HU2	HU2	Verbatim		None
	Initiating Condition: Seismic event greater than OBE levels.	Seismic event greater than OBE levels.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits)	(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by seismic monitor indication of ground acceleration greater than: • 0.06 g horizontal **OR** • 0.04 g vertical	Difference	Global Comment #9 & 12 Sentence structure altered from list to accommodate a single indication as defined by PBNP procedure NP 7.2.29. Intent and meaning of the EALs are not altered.	V26

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HU3	Recognition Category: HU3	HU3	Verbatim	Global Comment #14	None
	Initiating Condition: Hazardous event.	Hazardous event.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A tornado strike within the PROTECTED AREA.	(1) A tornado strike within the PROTECTED AREA.	Verbatim	Global Comment #12	None
	(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	Verbatim		None
	(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	Verbatim		None
	(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	Verbatim		None
	(5) (Site-specific list of natural or technological hazard events)	(5) Lake level greater than or equal to +8.0 ft. (Plant elevation)	Difference	Global Comment #9	V14
		(6) Pump bay level less than -15.0 ft.	Difference	Global Comment #9 Added as a 6th EAL versus a list in EAL #5 to maintain consistent format Intent and meaning of the EALs are not altered.	V14

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HU4	Recognition Category: HU4	HU4	Verbatim	Global Comment #14	None
	Initiating Condition: FIRE potentially degrading the level of safety of the plant.	FIRE potentially degrading the level of safety of the plant.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) a. A FIRE is NOT extinguished within 15-minutes of ANY of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND**	(1) a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND**	Difference	Global Comment #12 & 13	None
	b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas)	b. The FIRE is located within **ANY** Table H-1 plant rooms or areas:	Difference	Global Comment #8, 9, & 13	None
	(2) a. Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas) **AND**	(2) a. Receipt of a single fire alarm with no other indications of a FIRE. **AND** b. The FIRE is located within **ANY** Table H-1 plant rooms or areas except Containment in Modes 1 and 2 (see Note above): **AND**	Deviation	PBNP proposes to make EAL HU4.2 applicable to a single fire alarm in Containment during operation in Modes 3 and 4 only. Note added to IC that a *"Containment fire alarm is considered valid upon receipt of multiple zones (more than 1) on the FACP system (this note is applicable in Modes 1 and 2 only)."* Global Comment #8, 9 & 13	V35

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
	c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.	c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.			
	(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.	(2) A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication	Difference	Global Comment #9	None
	(4) A FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	(3) A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	Difference	Global Comment #9	V27
		Table H-1 Areas Control Room Containment PAB G05 building 13.8kV Building Cable Spreading Room Vital Switchgear Room AFW Pump Room G-01/02 Rooms EDG Building Service Water Pump Rooms Façade 85'	Difference	Global Comment #8 & 9 Basis revised to include clarification of Containment fire alarms, and to include NFPA-805 in the discussion of Appendix R basis for the EAL thresholds. Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HU7	Recognition Category: HU7	HU7	Verbatim	Global Comment #14	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO) UE.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO) UE.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	Verbatim	Global Comment #12	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HA1	Recognition Category: HA1	HA1	Verbatim	Global Comment #14	None
	Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.	Difference	Global Comment #9 & 12	None
	(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	Verbatim	Intent and meaning of the EALs are not altered.	

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="4">HA5</td><td>Recognition Category: HA5</td><td>HA5</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.</td><td>Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: All</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>(1) a. 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 of the following plant rooms or areas:
<table><tr><th colspan="2">Table H-2
SAFE OPS, S/D, C/D AREAS</th></tr><tr><th>Area/Building</th><th>MODE</th></tr><tr><td>U1 VCT Area</td><td>3 / 4 / 5</td></tr><tr><td>U2 VCT Area</td><td>3 /4 / 5</td></tr><tr><td>U1 Primary Sample area</td><td>3</td></tr><tr><td>U2 Primary Sample area</td><td>3</td></tr><tr><td>CCW HX Room</td><td>4 / 5</td></tr><tr><td>C-59 area</td><td>3 / 4 / 5</td></tr><tr><td>Pipeway 2, 8 ft. Elev.</td><td>3 / 4</td></tr><tr><td>Pipeway 3, 8 ft. Elev.</td><td>3 / 4</td></tr><tr><td>1/2B32 MCC Area</td><td>4</td></tr></table>AND
a. Entry into the room or area is prohibited or impeded.</td><td>Difference

Verbatim</td><td>Global Comment #8, 9 & 12

Intent and meaning of the EALs are not altered.</td><td>V28

None</td></tr>
</table>

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HA6	Recognition Category: HA6	HA6	Verbatim		None
	Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.	Control Room evacuation resulting in transfer of plant control to alternate locations.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	(1) An event has resulted in plant control being transferred from the Control Room to AOP local control stations.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HA7	Recognition Category: HA7	HA7	Verbatim	Global Comment #14	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HS1	Recognition Category: HS1	HS1	Verbatim	Global Comment #14	None
	Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.	HOSTILE ACTION within the PROTECTED AREA.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HS6	Recognition Category: HS6	HS6	Verbatim		None
	Initiating Condition: Inability to control a key safety function from outside the Control Room.	Inability to control a key safety function from outside the Control Room.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that (site specific number of) minutes has been exceeded, or will likely be exceeded.	**Note:** The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.		Global Comment #9 15 minutes used per Developers Note	None
	(1) a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	(1) a. An event has resulted in plant control being transferred from the Control Room to AOP local control stations.	Difference	Global Comment #9 & 12	None
	AND b. Control of **ANY** of the following key safety functions is not reestablished within (site-specific number of minutes). • Reactivity control • Core cooling [PWR] / RPV water level [BWR] • RCS heat removal	**AND** b. Control of **ANY** of the following key safety functions is not reestablished within 15 minutes. • Reactivity Control • Core Cooling • RCS Heat Removal	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HS7	Recognition Category: HS7	Recognition Category: HS7	Verbatim	Global Comment #14	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: ALL	Verbatim		None
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HG1	Recognition Category: HG1	HG1	Verbatim	Global Comment #14	None
	Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility.	HOSTILE ACTION resulting in loss of physical control of the facility.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.	Difference	Global Comment #9 & 12	None
	AND b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity control • Core cooling [PWR] / RPV water level [BWR] • RCS heat removal	**AND** b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity Control • Core Cooling • RCS Heat Removal	Difference	Global Comment #9	None
	OR 2. Damage to spent fuel has occurred or is IMMINENT.	**OR** 2. Damage to spent fuel has occurred or is IMMINENT.	Verbatim	Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
HG7	Recognition Category: HG7	HG7	Verbatim	Global Comment #14	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

SYSTEM MALFUNCTION ICS/EALS

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SU1	Recognition Category: SU1	SU1	Verbatim		None
	Initiating Condition: Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.	Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) Loss of **ALL** offsite AC power capability to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite AC power capability to 1(2)-A05 and 1(2)-A06 for 15 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="4">SU2</td><td>Recognition Category: SU2</td><td>SU3</td><td>Verbatim</td><td>Global Comment #14, Renumbered to align with other similar ICs</td><td>None</td></tr>
<tr><td>Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.</td><td>UNPLANNED loss of Control Room indications for 15 minutes or longer.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown</td><td>Operating Mode Applicability: 1, 2, 3, 4</td><td>Difference</td><td>Global Comment #10</td><td>None</td></tr>
<tr><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.

<table>
<tr><th>[BWR parameter list]</th><th>[PWR parameter list]</th></tr>
<tr><td>Reactor Power</td><td>Reactor Power</td></tr>
<tr><td>RPV Water Level</td><td>RCS Level</td></tr>
<tr><td>RPV Pressure</td><td>RCS Pressure</td></tr>
<tr><td>Primary Containment Pressure</td><td>In-Core/Core Exit Temperature</td></tr>
<tr><td>Suppression Pool Level</td><td>Levels in at least (site-specific number) two steam generators</td></tr>
<tr><td>Suppression Pool Temperature</td><td>Steam Generator Auxiliary or Emergency Feed Water Flow</td></tr>
</table>
</td><td>(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 / Pressurizer Level
• RCS / Pressurizer Pressure
• Core Exit Temperature
• Level in at least one steam generator
• Steam Generator Auxiliary Feed Water Flow</td><td>Difference

Difference</td><td>Global Comment #9 & 12

Global Comment #9

Intent and meaning of the EALs are not altered.</td><td>None

None</td></tr>
</table>

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SU3	Recognition Category: SU3	SU4	Verbatim	Renumbered to align with other similar ICs	None
	Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.	Reactor coolant activity greater than Technical Specification allowable limits.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) (Site-specific radiation monitor) reading greater than (site-specific value).	(1) Failed Fuel Monitor 1(2)-RE-109 reading greater than 750 mR/hr	Difference	Global Comment #9 & 12	V30
	(2) Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	(2) Sample analysis indicates that a RCS Specific Activity value is greater than an allowable limit specified in Technical Specifications as indicated by ANY of the following conditions: a. Dose Equivalent I-131 greater than 50 µCi/gm OR b. Dose Equivalent I-131 greater than 0.5 µCi/gm but less than or equal to 50 µCi/gm for greater than 48 hours OR c. Dose Equivalent Xe-133 greater than 300 µCi/gm for greater than 48 hours	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V31

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SU4	Recognition Category: SU4	SU5	Verbatim	Global Comment #14, Renumbered to align with other similar ICs	None
	Initiating Condition: RCS leakage for 15 minutes or longer.	RCS leakage for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.	(1) RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.	Difference	Global Comment #9 & 12	None
	(2) RCS identified leakage greater than (site-specific value) for 15 minutes or longer.	(2) RCS identified leakage greater than 25 gpm for 15 minutes or longer.	Difference	Global Comment #9	V32
	(3) Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.	(3) Leakage from the RCS to a location outside containment, or Steam Generator tube leakage, greater than 25 gpm for 15 minutes or longer.	Difference	Global Comment #9 Added specific language about SG tube leakage to the EAL at the request of operators Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SU5	Recognition Category: SU5	SU6	Verbatim	Global Comment #14, Renumbered to align with other similar ICs	None
	Initiating Condition: Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor.	Automatic or manual trip fails to shutdown the reactor.	Difference	Global Comment #4	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1	Difference	Global Comment #10	None
	(1) a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic trip did not shutdown the reactor.	Difference	Global Comment #4 & 12	None
	AND b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	**AND** b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	Verbatim		None
	(2) a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor.	(2) a. A manual trip did not shutdown the reactor.	Difference	Global Comment #4	None
	AND b. EITHER of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	**AND** b. EITHER of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	Verbatim		None
	OR 2. A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.	**OR** 2. A subsequent automatic trip is successful in shutting down the reactor.	Difference	Global Comment #4 Added Note to basis to define PBNP "at the reactor control consoles" as meaning the reactor trip pushbuttons on the following panels: • Unit 1 on panels 1C04 and C01 • Unit 2 on panels 2C04 and C02 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SU6	Recognition Category: SU6	SU7	Verbatim	Renumbered to align with other similar ICs	None
	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Loss of all onsite or offsite communications capabilities.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) Loss of ALL of the following Onsite communication methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following Onsite communication methods: • Plant Public Address System (Gai-Tronics) • Commercial Phones • PBX Phones • Security Radio • Portable Radios	Difference	Global Comment #9, 12 & 13	None
	(2) Loss of ALL of the following ORO communications methods: (site-specific list of communications methods)	(2) Loss of ALL of the following offsite response organization communications methods: • Nuclear Accident Reporting System (NARS) • Commercial Phones • PBX Phones • Satellite Phones • Manitowoc County Sheriff's Department Radio	Difference	Global Comment #9 & 13	None
	(3) Loss of ALL of the following NRC communications methods: (site-specific list of communications methods)	(2) Loss of **ALL** of the following NRC communications methods: • FTS Phone System • Commercial Phones • PBX Phones • Satellite Phones	Difference	Global Comment #9 & 13 Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SU7	Recognition Category: SU7	SU8	Verbatim	Renumbered to align with other similar ICs	None
	Initiating Condition: Failure to isolate containment or loss of containment pressure control. [PWR]	Failure to isolate containment or loss of containment pressure control.	Difference	Global Comment #4	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) a. Failure of containment to isolate when required by an actuation signal.	(1) a. Failure of containment to isolate when required by an actuation signal.	Verbatim	Global Comment #12	None
	AND b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal.	**AND** b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal.	Verbatim		None
	(2) a. Containment pressure greater than (site-specific pressure).	(2) a. Containment pressure greater than 25 psig.	Difference	Global Comment #9	V25
	AND b. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.	**AND** b. Less than one full train of Containment Cooling System equipment is operating per design for 15 minutes or longer.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SA1	Recognition Category: SA1	SA1	Verbatim	Global Comment #14	None
	Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. AND	(1) a. AC power capability to 1(2)-A05 **AND** 1(2)-A06 is reduced to a single power source for 15 minutes or longer. **AND**	Difference	Global Comment #9, 12 & 13	None
	b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.	b. Any additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.	Difference	Global Comment #13 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>PBNP</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="5">SA2</td><td>Recognition Category: SA2</td><td>SA3</td><td>Verbatim</td><td>Global Comment #14, Renumbered to align with other similar ICs</td><td>None</td></tr>
<tr><td>Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</td><td>UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown</td><td>Operating Mode Applicability: 1, 2, 3, 4</td><td>Difference</td><td>Global Comment #10</td><td>None</td></tr>
<tr><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.</td><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.</td><td>Verbatim</td><td>Global Comment #12</td><td>None</td></tr>
<tr><td><table>
<tr><td>[BWR parameter list]</td><td>[PWR parameter list]</td></tr>
<tr><td>Reactor Power</td><td>Reactor Power</td></tr>
<tr><td>RPV Water Level</td><td>RCS Level</td></tr>
<tr><td>RPV Pressure</td><td>RCS Pressure</td></tr>
<tr><td>Primary Containment Pressure</td><td>In-Core/Core Exit Temperature</td></tr>
<tr><td>Suppression Pool Level</td><td>Levels in at least (site-specific number) steam generators</td></tr>
<tr><td>Suppression Pool Temperature</td><td>Steam Generator Auxiliary or Emergency Feed Water Flow</td></tr>
</table></td><td>• Reactor Power
• RCS / Pressurizer Level
• RCS / Pressurizer Pressure
• Core Exit Temperature
• Levels in at least one steam generator
• Steam Generator Auxiliary Feed Water Flow</td><td>Difference</td><td>Global Comment #4, 8 & 9</td><td>None</td></tr>
</table>

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
	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]	**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	Difference	Global Comment #4, 9 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SA5	Recognition Category: SA5	SA6	Verbatim	Renumbered to align with other similar ICs	None
	Initiating Condition: Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	Automatic or manual trip fails to shutdown the reactor, and subsequent manual actions taken in the Control room are not successful in shutting down the reactor.	Difference	Global Comment #4 & 9	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1	Difference	Global Comment #10	None
	(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic or manual trip did not shutdown the reactor.	Difference	Global Comment #4, 9 & 12	None
	AND b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	**AND** b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	Verbatim	Added Note to basis to define PBNP "at the reactor control consoles" as meaning the reactor trip pushbuttons on the following panels: • Unit 1 on panels 1C04 and C01 • Unit 2 on panels 2C04 and C02 Intent and meaning of the EALs are not altered.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SA9	Recognition Category: SA9	SA9	Verbatim	Global Comment #14	None
	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) a. The occurrence of ANY of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager	(1) a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Lake level greater than or equal to +9.0 ft. (Plant elevation) • Pump bay level less than -19.0 ft. • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director	Difference Difference	Global Comment #12 & 13 Global Comment #8 & 9	None V14

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SA9 (cont.)	**AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.	**AND** b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode. **AND**	Deviation	Adopted the revised EAL structure and wording provided in proposed EAL FAQ 2016-02.	V15
	OR 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	2. **EITHER** of the following: • Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or • The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.	Deviation	Adopted the revised EAL structure and wording provided in proposed EAL FAQ 2016-02. Intent and meaning of the EALs are not altered.	V15

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SS1	Recognition Category: SS1	SS1	Verbatim	Global Comment #14	None
	Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite and **ALL** onsite AC power to 1(2)-A05 and 1(2)-A06 for 15 minutes or longer.	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SS5	Recognition Category: SS5	SS6	Verbatim	Renumbered to align with other similar ICs	None
	Initiating Condition: Inability to shutdown the reactor causing a challenge to (core cooling [PWR] / RPV water level [BWR]) or RCS heat removal.	Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal.	Difference	Global Comment #4	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1	Difference	Global Comment #10	None
	(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic or manual trip did not shutdown the reactor.	Difference	Global Comment #4, 9 & 12	None
	AND b. All manual actions to shutdown the reactor have been unsuccessful.	**AND** b. All manual actions to shutdown the reactor have been unsuccessful.	Verbatim		None
	AND c. EITHER of the following conditions exist: • (Site-specific indication of an inability to adequately remove heat from the core) • (Site-specific indication of an inability to adequately remove heat from the RCS)	**AND** c. EITHER of the following conditions exist: • Conditions requiring entry into Core Cooling – Red path (CSP-C.1) are met. • Conditions requiring entry into Heat Sink – Red path (CSP-H.1) are met.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V19 & V20

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SS8	Recognition Category: SS8	SS2	Verbatim	Global Comment #14; renumbered to align with other emergency power source ICs	None
	Initiating Condition: Loss of all Vital DC power for 15 minutes or longer.	Loss of all Vital DC power for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) Indicated voltage is less than (site-specific bus voltage value) on **ALL** (site-specific Vital DC busses) for 15 minutes or longer.	(1) Indicated voltage is less than 115 VDC on **ALL** Vital DC busses D-01, D-02, D-03, and D-04 for 15 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V33

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SG1	Recognition Category: SG1	SG1	Verbatim	Global Comment #14	None
	Initiating Condition: Prolonged loss of all offsite and all onsite AC power to emergency buses.	Prolonged loss of all offsite and all onsite AC power to emergency buses.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) a. Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses).	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to 1(2)-A05 and 1(2)-A06.	Difference	Global Comment #9 & 13	None
	AND b. EITHER of the following: • Restoration of at least one AC emergency bus in less than (site-specific hours) is not likely.	**AND** b. **EITHER** of the following: • Restoration of at least one AC emergency bus in less than 4 hours is not likely.	Difference	Global Comment #9 & 13	V34
	• (Site-specific indication of an inability to adequately remove heat from the core)	• Conditions requiring entry into Core Cooling – Red Path (CSP-C1) are met.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V19

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
SG8	Recognition Category: SG8	SG2	Verbatim	Global Comment #14; renumbered to align with other emergency power source ICs	None
	Initiating Condition: Loss of all AC and Vital DC power sources for 15 minutes or longer.	Loss of all AC and Vital DC power sources for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	Global Comment #10	None
	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to 1(2)-A05 and 1(2)-A06 for 15 minutes or longer.	Difference	Global Comment #9 & 12	None
	AND b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer.	**AND** b. Indicated voltage is less than 115 VDC on **ALL** Vital DC busses D-01, D-02, D-03, and D-04 for 15 minutes or longer.	Difference	Global Comment #9 & 13 Intent and meaning of the EALs are not altered.	V33

APPENDIX A – ACRONYMS AND ABBREVIATIONS

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS	AC.......Alternating Current	AC.......Alternating Current	Verbatim		N/A
	AOP......Abnormal Operating Procedure	AOP......Abnormal Operating Procedure	Verbatim		N/A
	APRM...Average Power Range Meter		Difference	Not used	N/A
	ATWS...Anticipated Transient Without Scram	ATWS...Anticipated Transient Without Scram	Verbatim		N/A
	B&W....Babcock and Wilcox		Difference	Not used	N/A
	BIIT......Boron Injection Initiating Temperature		Difference	Not used	N/A
	BWR....Boiling Water Reactor		Difference	Not used	N/A
	CDE......Committed Dose Equivalent	CDE......Committed Dose Equivalent	Verbatim		N/A
	CFR......Code of Federal Regulations	CFR......Code of Federal Regulations	Verbatim		N/A
	CTMT/CNMT...Containment	CNMT...Containment	Difference	CTMT not used	N/A
	CSF......Critical Safety Function	CSF......Critical Safety Function	Verbatim		N/A
	CSFST...Critical Safety Function Status Tree	CSFST...Critical Safety Function Status Tree	Verbatim		N/A
	DBA......Design Basis Accident		Difference	Not used	N/A
	DC........Direct Current	DC........Direct Current	Verbatim		N/A
	EAL.......Emergency Action Level	EAL.......Emergency Action Level	Verbatim		N/A
	ECCS....Emergency Core Cooling System	ECCS....Emergency Core Cooling System	Verbatim		N/A
	ECL.......Emergency Classification Level	ECL.......Emergency Classification Level	Verbatim		N/A
	EOF......Emergency Operations Facility	EOF......Emergency Operations Facility	Verbatim		N/A
	EOP......Emergency Operating Procedure	EOP......Emergency Operating Procedure	Verbatim		N/A
	EPA......Environmental Protection Agency	EPA......Environmental Protection Agency	Verbatim		N/A
	EPG.....Emergency Procedure Guideline		Difference	Not used	N/A
	EPIP.....Emergency Planning Implementing Procedure		Difference	Not used	N/A

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	EPR......Evolutionary Power Reactor		Difference	Not used	N/A
	EPRI.....Electric Power Research Institute		Difference	Not used	N/A
	ERG.....Emergency Response Guideline		Difference	Not used	N/A
	FEMA...Federal Emergency Management Agency	FEMA...Federal Emergency Management Agency	Verbatim		N/A
	FSAR....Final Safety Analysis Report		Difference	Not used	N/A
	GE........General Emergency	GE........General Emergency	Verbatim		N/A
	HCTL....Heat Capacity Temperature Limit		Difference	Not used	N/A
	HPCI.....High Pressure Coolant Injection		Difference	Not used	N/A
	HSI........Human System Interface		Difference	Not used	N/A
	IC..........Initiating Condition	IC..........Initiating Condition	Verbatim		N/A
	ID.........Inside Diameter	ID.........Inside Diameter	Verbatim		N/A
	IPEEE...Individual Plant Examination of External Events (Generic Letter 88-20)		Difference	Not used	N/A
	ISFSI....Independent Spent Fuel Storage Installation	ISFSI....Independent Spent Fuel Storage Installation	Verbatim		N/A
	Keff.....Effective Neutron Multiplication Factor	Keff.....Effective Neutron Multiplication Factor	Verbatim		N/A
	LCO.....Limited Condition of Operation	LCO.....Limited Condition of Operation	Verbatim		N/A
	LOCA...Loss of Coolant Accident	LOCA...Loss of Coolant Accident	Verbatim		N/A
	MCR....Main Control Room		Verbatim		N/A
	MSIV...Main Steam Isolation Valve		Difference	Not used	N/A
	MSL.....Main Stem Line		Difference	Not used	N/A
	mR, mRem, mrem, mREM....milli-Roentgen Equivalent Man	mR, mRem, mrem, mREM....milli-Roentgen Equivalent Man	Verbatim		N/A
	MW.....Megawatt	MW.....Megawatt	Verbatim		N/A
	NEI.......Nuclear Energy Institute	NEI.......Nuclear Energy Institute	Verbatim		N/A
	NPP......Nuclear Power Plant		Difference	Not used	N/A

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	NRC.....Nuclear Regulatory Agency	NRC.....Nuclear Regulatory Agency	Verbatim		N/A
	NSSS....Nuclear Steam Supply System		Difference	Not used	N/A
	NORAD...North American Aerospace Defense Command	NORAD...North American Aerospace Defense Command			N/A
	(NO)UE...(Notification of) Unusual Event	(NO)UE...(Notification of) Unusual Event	Verbatim		N/A
	NUMARC....Nuclear Management and Resources Council	NUMARC....Nuclear Management and Resources Council	Verbatim		N/A
	OBE.....Operating Basis Earthquake	OBE.....Operating Basis Earthquake	Verbatim		N/A
	OCA.....Owner Controlled Area	OCA.....Owner Controlled Area	Verbatim		N/A
	ODCM/ODAM....Offsite Dose Calculation (Assessment) Manual	ODCM...Offsite Dose Calculation Manual	Difference	PBNP uses ODCM	N/A
	ORO.....Offsite Response Organization	ORO.....Offsite Response Organization	Verbatim		N/A
	PA.........Protected Area	PA.........Protected Area	Verbatim		N/A
	PACS....Priority Information and Control System		Difference	Not used	N/A
	PAG......Protective Action Guideline	PAG......Protective Action Guideline	Verbatim		N/A
	PICS.....Process Information and Control System		Difference	Not used	N/A
	PRA/PSA...Probabilistic Risk Assessment/Probabilistic Safety Assessment	PRA/PSA...Probabilistic Risk Assessment/Probabilistic Safety Assessment	Verbatim		N/A
	PWR....Pressurized Water Reactor	PWR....Pressurized Water Reactor	Verbatim		N/A
	PS.........Protection System		Difference	Not used	N/A
	PSIG....Pounds per Square Inch	PSIG....Pounds per Square Inch	Verbatim		N/A
	R..........Roentgen	R..........Roentgen	Verbatim		N/A
	RCC....Reactor Control Console		Difference	Not used	N/A
	RCIC...Reactor Core Isolation Cooling		Difference	Not used	N/A

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	RCS.....Reactor Coolant System	RCS.....Reactor Coolant System	Verbatim		N/A
	Rem, rem, REM...Roentgen Equivalent Man	Rem, rem, REM...Roentgen Equivalent Man	Verbatim		N/A
	RETS....Radiological Effluent Technical Specifications		Difference	Not used	N/A
	RPS......Reactor Protection System	RPS......Reactor Protection System	Verbatim		N/A
	RPV......Reactor Pressure Vessel	RPV......Reactor Pressure Vessel	Verbatim		N/A
	RVLIS...Reactor Vessel Level Instrumentation System	RVLIS...Reactor Vessel Level Instrumentation System	Verbatim		N/A
	RWCU...Reactor Water Cleanup		Difference	Not used	N/A
	SAR.......Safety Analysis Report		Difference	Not used	N/A
	SAS........Safety Automation System		Difference	Not used	N/A
	SBO.......Station Blackout		Difference	Not used	N/A
	SCBA.....Self-Contained Breathing Apparatus	SCBA.....Self-Contained Breathing Apparatus	Verbatim		N/A
	SG..........Steam Generator	SG..........Steam Generator	Verbatim		N/A
	SI...........Safety Injection	SI...........Safety Injection	Verbatim		N/A
	SICS......Safety Information Control System		Difference	Not used	N/A
	SPDS.....Safety Parameter Display System	SPDS.....Safety Parameter Display System	Verbatim		N/A
	SRO.......Senior Reactor Operator		Difference	Not used	N/A
	TEDE.....Total Effective Dose Equivalent	TEDE.....Total Effective Dose Equivalent	Verbatim		N/A
	TOAF.....Top of Active Fuel	TOAF.....Top of Active Fuel	Verbatim		N/A
	TSC........Technical Support System	TSC........Technical Support System	Verbatim		N/A
	-	UFSAR....Final Safety Analysis Report	Difference	Used in Section 3.1	N/A
	WOG.....Westinghouse Owners Group	WOG.....Westinghouse Owners Group	Verbatim		N/A

APPENDIX B - DEFINITIONS

PBNP DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	Alert: Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.	Alert: Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.	Verbatim		None
	General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.	General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.	Verbatim		None
	Notification of Unusual Event: Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	Unusual Event: Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	Difference	See Global Comment #3	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.	Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.	Verbatim		None
	Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.	Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.	Verbatim		None
	Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: • Notification of Unusual Event (NOUE) • Alert • Site Area Emergency (SAE) • General Emergency (GE)	Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: • Unusual Event (UE) • Alert • Site Area Emergency (SAE) • General Emergency (GE)	Difference	See Global Comment #3	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.	Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.	Verbatim		None
	Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.	Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.	Verbatim		None
	CONFINEMENT BOUNDARY: (Insert a site-specific definition for this term.) **Developer Note** – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.	CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. This corresponds to the pressure boundary for the Dry Shielded Canister (DSC) in NUHOMS 32PT casks and the Multi-assembly Sealed Basket (MSB) for the VSC-24 Storage System.	Difference	Removed developer notes and added site-specific language.	None
	CONTAINMENT CLOSURE: (Insert a site-specific definition for this term.) **Developer Note** – The procedurally defined conditions or actions taken to secure containment (primary or secondary for BWR) and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.	CONTAINMENT CLOSURE: Is the action taken to secure containment and its assorted structures, systems and components as a functional barrier to fission product release under existing plant conditions. CONTAINMENT CLOSURE is initiated per the SEPs or Shift Manager direction if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. CONTAINMENT CLOSURE requires that, upon a loss of decay heat removal, any open penetration which is listed on CL 1E, Containment Closure Checklist, must be closed or capable of being closed prior to RCS bulk boiling. This checklist is maintained any time that the RCS is <200°F and CONTAINMENT operability is not maintained.	Difference	Removed developer notes and added existing definition from present EALs.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.	EXPLOSION: A rapid, violent, and catastrophic failure of a piece of equipment due to combustion, chemical reaction, or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.	Verbatim		None
	FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized. Developer Note – This term is applicable to PWRs only.	FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.	Difference	Removed developer note.	None
	FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.	FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.	Verbatim		None
	HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.	HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.	Verbatim		None

IC	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
	HOSTILE ACTION: An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).	HOSTILE ACTION: An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).	Difference	Spelled out 'NPP' in 2 places	None
	HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.	HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.	Verbatim		None
	IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.	IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.	Verbatim		None
	INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.	INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.	Verbatim		None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	NORMAL LEVELS: As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.		Difference	Term not used in this EAL scheme	None
	OWNER CONTROLLED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the site property owned by, or otherwise under the control of, the licensee. In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan.	OWNER CONTROLLED AREA: The site property owned by or otherwise under the control of the licensee.	Difference	Definition from developer notes used. Developer Notes deleted.	None
	PROJECTILE: An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.	PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.	Difference	Spelled out 'NPP'	None
	PROTECTED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the area under continuous access monitoring and control, and armed protection as described in the site Security Plan.	PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.	Difference	Definition from developer notes used. Developer Notes deleted.	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	REFUELING PATHWAY: (Insert a site-specific definition for this term.) **Developer Note** – This description should include all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.	REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool, and fuel transfer canal.	Difference	PBNP-specific definition supplied. Developer Notes deleted.	None
	RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection. **Developer Note** – This term is applicable to PWRs only.	RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.	Difference	Removed developer notes.	None
	SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related. **Developer Note** – This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.	SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.	Difference	Removed developer notes and clarified last sentence.	None
	SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.	SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.	Verbatim		None
		SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.	Difference	Defined term from ODCM needed for several EALs	None

PBNP DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
	UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.	UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.	Verbatim		None
APPENDIX B - DEFINITIONS	UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.	UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.	Verbatim		N/A
	VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.	VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.	Deviation	Updated to reflect wording and guidance of proposed EAL FAQ 2016-02. The updated wording clarifies damage assessment meriting an ALERT declaration as used in ICs using this definition (CA6 and SA9).	V15

APPENDIX C - Permanently Defueled ICs/EALs

PBNP DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	PBNP	Change	Justification	Validation #
Appendix C – Permanently Defueled ICs/EALs	Appendix C - Permanently Defueled ICs/EALs	Not used at PBNP	Difference	Not applicable to PBNP	None

ATTACHMENT 4

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 286, ADOPTION OF EMERGENCY ACTION LEVEL SCHEME PURSUANT TO NEI 99-01 REVISION 6, "DEVELOPMENT OF EMERGENCY ACTION LEVELS FOR NON-PASSIVE REACTORS"

UPDATED SUPPORTING TECHNICAL INFORMATION

189 pages follow

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[(b)]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

licensee to promptly declare the emergency condition as soon as possible following the identification of the appropriate ECL. As used here, "promptly" means the next available opportunity unimpeded by activities not related to the emergency declaration, unless such activities are necessary for protecting health and safety. (See Paragraph 8 of this section.)

6. Consistent with the NRC's position that emergency declarations are made promptly, the final rule states that the 15-minute criterion not be construed as a grace period in which a licensee may attempt to restore plant conditions to avoid declaring an EAL that has already been exceeded. This statement does not preclude licensees from acting to correct or mitigate an off-normal condition, but once an EAL has been recognized as being exceeded, the emergency declaration shall be made promptly without waiting for the 15-minute period to elapse. This is particularly the case when the EAL threshold is exceeded based on occurrence of a condition, rather than the duration of a condition.

7. For EAL thresholds that specify a duration of the off-normal condition, the NRC expects that the emergency declaration process run concurrently with the specified threshold duration. Once the off-normal condition has existed for the duration specified in the EAL, no further effort on this declaration is necessary—the EAL has been exceeded. Consider as an example, the EAL "fire which is not extinguished within 15 minutes of detection." On receipt of a fire alarm, the plant fire brigade is dispatched to the scene to begin fire suppression efforts.

   - If the fire brigade reports that the fire can be extinguished before the specified duration, the emergency declaration is placed on hold while firefighting activities continue. If the fire brigade is successful in extinguishing the fire within the specified duration from detection, no emergency declaration is warranted based on that EAL.

   - If the fire is still burning after the specified duration has elapsed, the EAL is exceeded, no further assessment is necessary, and the emergency declaration would be made promptly. As used here, "promptly" means at the first available opportunity (e.g., if the Shift Manager is receiving an update from the fire brigade at the 15-minute mark, it is expected that the declaration will occur as the next action after the call ends).

   - If, for example, the fire brigade notifies the shift supervision 5 minutes after detection that the brigade itself cannot extinguish the fire such that the EAL will be met imminently and cannot be avoided, the NRC would not consider it a violation of the licensee's emergency plan to declare the event before the EAL is met (e.g., the 15-minute duration has elapsed). While a prompt declaration would be beneficial to public health and safety and is encouraged, it is not required by regulation.

   - In all of the above, the fire duration is measured from the time the alarm, indication, or report was first received by the plant operators. Validation or confirmation establishes that the fire started as early as the time of the alarm, indication, or report.

The monitor readings that correspond to the AU1 initiating condition (2X ODCM Alarm Setpoints) under NEI 99-01 Revision 4 are listed below:

Monitor	Reading
RE-315, Auxiliary Building Exhaust Low Range Gas	5.4E-03 µCi/cc
RE-317, Auxiliary Building Exhaust Low Range Gas	5.4E-03 µCi/cc
1(2)RE-307 Containment Purge Exhaust Mid-Range Gas, with only containment purge in operation (25,000 cfm)	1.4E-02 µCi/cc
2RE-307, Containment Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	9.4E-03 µCi/cc
2RE-305, Containment Purge Exhaust Low-Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	9.4E-03 µCi/cc
2RE-307, Containment Purge Exhaust Mid-Range Gas with only GS building ventilation in operation (13,000 cfm).	2.8E-02 µCi/cc
2RE-307, Containment Purge Exhaust Mid-Range Gas with only forced vent of containment (35 cfm).	1.0E+01 µCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with only forced vent of containment (35 cfm).	1.0E+01 µCi/cc
RE-325, Drumming Area Exhaust Low-Range Gas	8.4E-03 µCi/cc
RE-327, Drumming Area Exhaust Mid-Range Gas	8.4E-03 µCi/cc

TABLE 9-1
LIQUID EFFLUENT PATHWAYS

LIQUID EFFLUENT PATHWAY	PATHWAY MONITOR[3]		DISCHARGE FLOWRATE (GPM)	CALCULATED DEFAULT SETPOINT[1] (µCi/cc)
Recirculation Water	None	1 pump, either unit	243,000	N/A
		2 pumps, either unit	394,000	N/A
		1 pump, each unit	484,000	N/A
		1 pump, one unit & 2 pumps, other unit	619,000	N/A
		2 pumps, each unit	744,000	N/A
Service Water Return (normal cool down per pump)	1(2)RE-229	2 pumps @ 7500 gpm	15,000	
		3 pumps @ 6300 gpm	18,900	
		4 pumps @ 5100 gpm	20,400	
		5 pumps @ 4300 gpm	21,500	
		6 pumps @ 3700 gpm	22,200	1.14E-03
Steam Generator Blowdown	1(2)RE-219* & 1 (2)RE-222	Max Flow Rate	200	1.26E-01
Waste Water Effluent[2]	RE-230	Max Flow Rate (both filter skids running in parallel)	700	1.22E-03
Spent Fuel Pool	RE-220*	Max Flow Rate	700	3.61E-02
Waste Distillate & Condensate Storage Tank Discharge	RE-218* & RE-223*	Max Flow Rate	100	2.53E-01
Containment Fan Cooler Return	1(2)RE-216*	Max Flow Rate (per Containment)	4000	6.32E-03

NOTE 1: **Setpoints except for RE-230 are based on 10x the MEC values listed in 10CFR20, Appendix B, Table 2, Column 2. PBNP TS Section 5.5.4.b allows concentrations of radioactive material released to unrestricted areas to be 10x the MEC values.**

NOTE 2: **RE-230 setpoint explanation can be found in section 9.1, Default Monitor Setpoints.**

NOTE 3: **Monitors marked with an asterisk (*) have a calculated default alarm setpoint above the monitors fail high or saturation level. See section 9.1, High Alarm or Trip Setpoint Guidelines for further explanation.**

2 times 1.14E-03 = 2.28E-03

Value rounded to 2.3E-03 for use in EAL RU1.1

Document Information:

Calculation (Doc) No: 2013-0018	Controlled Documents Revision: 2
Title: RADIOLOGICAL EFFLUENT INITIATING CONDITION VALUES FOR EMERGENCY ACTION LEVEL RS1 AND RG1	
Type: CALC Sub-Type: CALC Discipline: Radiological	
Facility: **PBNP**	Unit: 0
Safety Class: ☐ SR ☒ Quality Related ☐ Non-Nuclear Safety ☐ Important to Safety ☐ Not Important to Safety	
Special Codes: ☐ Safeguards ☐ Proprietary	
Vendor Doc No: N/A	Vendor Name or Code:N/A
Executive Summary (optional): Expand scope to include EAL ICs for Unusual Events and Alerts, and for both Revision 4 and Revision 6 of NEI 99-01. Delete calculation of setpoints based on a 4 hour assumed release duration (make consistent with NEI 99-01 assumption of 1 hour). Incorporate inputs from ODCM Revision 19.	

Review and Approval:

Associated EC Number: 288117 AR/ Other Document Number: 02179019		EC Revision: ~~0~~ 1 /ck. 2/20/17
Description of Calculation Revision: Update		EC Document Revision: 2
Prepared by: [signature] (signature)	T. C. Kendall (print name)	Date: 2/16/2017
Reviewed by: [signature] (signature)	Carl Onesti (print name)	Date: 2/16/2017
Type of Review: ☒ Design Verification ☐ Review ☐ Owner Acceptance Review		
Method Used (For DV Only): ☐ Design Review ☐ Alternate Calculation		
Approved by: [signature] (signature)	Jane Marean 2/20/17 (print name)	Date: 2/20/17

Results and Conclusions:

The results were calculated using a source term that is consistent with the ODCM and the guidance of NEI 99-01. The results are applicable for a reactor thermal power operating level of 1800 MWth and are applicable generally to other thermal power operating levels due to the use of activity fractional data and not absolute activity values.

The monitor readings that correspond to the AG1 initiating condition value of 1 rem TEDE or 5 rem Thyroid CDE are listed below.

Monitor	Reading
1(2)RE-309, Containment Purge Exhaust High Range Gas with only containment purge in operation (25,000 cfm)	600 μCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with both purge and GS Building in operation (38,000 cfm)	400 μCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with only GS Building in operation (13,000 cfm)	1200 μCi/cc
RE-317, Auxiliary Building Exhaust Mid-Range Gas	100 μCi/cc
RE-319, Auxiliary Building Exhaust High Range Gas	100 μCi/cc
RE-327, Drumming Area Exhaust Mid-Range Gas	Off scale high*
1(2) RE-231, Steam Line 1A(2A); 1(2)RE-232, Steam Line 1(A)2(A)	
Atmospheric Dump Valve (ADV) release	19 μCi/cc
Main Steam Safety Valve (MSSV) release	7.4 μCi/cc

*The upper range limit on R-327 is 100 μCi/cc, so this instrument would be over-ranged by the nominal EAL IC of 160 μCi/cc. However, this is close enough to the upper range of the instrument that it is judged appropriate to, in the absence of real-time dose projection capability and no other indications, to make a classification declaration based on a valid off-scale high reading of this instrument.

EAL Developer guidance from 99-01:

It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

Since the upper range of RE-327 is 100 μCi/cc and this instrument would be severely over-ranged by the calculated value of 160 μCi/cc (160% of range), this monitor will not be included in this EAL.

The monitor readings that correspond to the AS1 initiating condition value of 100 mrem TEDE or 5 rem Thyroid CDE are listed below for a one hour release duration.

Monitor	Reading
1(2)RE-307 Containment Purge Exhaust Mid-Range Gas, with only containment purge in operation (25,000 cfm)	60 µCi/cc
1(2)RE-309 Containment Purge Exhaust High-Range Gas, with only containment purge in operation (25,000 cfm)	60 µCi/cc
2RE-307, Containment Purge Exhaust Mid Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	40 µCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	40 µCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with only GS building ventilation in operation (13,000 cfm).	120 µCi/cc
RE-317, Auxiliary Building Exhaust Mid-Range Gas	10 µCi/cc
RE-319, Auxiliary Building Exhaust High Range Gas	10 µCi/cc
RE-327, Drumming Area Exhaust Mid-Range Gas	16 µCi/cc
1(2) RE-231, Steam Line 1A(2A); 1(2)RE-232, Steam Line 1(A)2(A)	
Atmospheric Dump Valve (ADV) release	1.9 µCi/cc
Main Steam Safety Valve (MSSV) release	0.74 µCi/cc

The monitor readings that correspond to the AA1 initiating condition (1/100 of AG1) under NEI 99-01 Revision 6 are listed below. ***These will require submittal and approval of a License Amendment Request to implement.***

Monitor	Reading
1(2)RE-307 Containment Purge Exhaust Mid-Range Gas, with only containment purge in operation (25,000 cfm)	6.0 μCi/cc
1(2)RE-309 Containment Purge Exhaust High-Range Gas, with only containment purge in operation (25,000 cfm)	6.0 μCi/cc
2RE-307, Containment Purge Exhaust Mid Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	4.0 μCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	4.0 μCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with only GS building ventilation in operation (13,000 cfm).	12 μCi/cc
RE-317, Auxiliary Building Exhaust Mid-Range Gas	1.0 μCi/cc
RE-319, Auxiliary Building Exhaust High Range Gas	1.0 μCi/cc
RE-327, Drumming Area Exhaust Mid-Range Gas	1.6 μCi/cc
1(2) RE-231, Steam Line 1A(2A); 1(2)RE-232, Steam Line 1(A)2(A)	
Atmospheric Dump Valve (ADV) release	0.19 μCi/cc
Main Steam Safety Valve (MSSV) release	0.074 μCi/cc

The monitor readings that correspond to the AA1 initiating condition (200X ODCM Alarm Setpoints) under NEI 99-01 Revision 4 are listed below:

Monitor	Reading
RE-315, Auxiliary Building Exhaust Low Range Gas	5.4E-01 μCi/cc
RE-317, Auxiliary Building Exhaust Low Range Gas	5.4E-01 μCi/cc
1(2)RE-307 Containment Purge Exhaust Mid-Range Gas, with only containment purge in operation (25,000 cfm)	1.4E+00 μCi/cc
1(2)RE-309 Containment Purge Exhaust High-Range Gas, with only containment purge in operation (25,000 cfm)	1.4E+00 μCi/cc
2RE-307, Containment Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	9.4E-01 μCi/cc
2RE-307, Containment Purge Exhaust Mid-Range Gas with only GS building ventilation in operation (13,000 cfm).	2.8E+00 μCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with only GS building ventilation in operation (13,000 cfm).	2.8E+00 μCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with only forced vent of containment (35 cfm).	1.0E+03 μCi/cc
RE-327, Drumming Area Exhaust Mid-Range Gas	8.4E-01 μCi/cc

The monitor readings that correspond to the AU1 initiating condition (2X ODCM Alarm Setpoints) under NEI 99-01 Revision 4 are listed below:

Monitor	Reading
RE-315, Auxiliary Building Exhaust Low Range Gas	5.4E-03 μCi/cc
RE-317, Auxiliary Building Exhaust Low Range Gas	5.4E-03 μCi/cc
1(2)RE-307 Containment Purge Exhaust Mid-Range Gas, with only containment purge in operation (25,000 cfm)	1.4E-02 μCi/cc
2RE-307, Containment Purge Exhaust Mid-Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	9.4E-03 μCi/cc
2RE-305, Containment Purge Exhaust Low-Range Gas with both purge and GS building ventilation in operation (38,000 cfm).	9.4E-03 μCi/cc
2RE-307, Containment Purge Exhaust Mid-Range Gas with only GS building ventilation in operation (13,000 cfm).	2.8E-02 μCi/cc
2RE-307, Containment Purge Exhaust Mid-Range Gas with only forced vent of containment (35 cfm).	1.0E+01 μCi/cc
2RE-309, Containment Purge Exhaust High Range Gas with only forced vent of containment (35 cfm).	1.0E+01 μCi/cc
RE-325, Drumming Area Exhaust Low-Range Gas	8.4E-03 μCi/cc
RE-327, Drumming Area Exhaust Mid-Range Gas	8.4E-03 μCi/cc

1. Purpose and Scope

The purpose of this calculation is to determine the expected dose rates on radiation monitors RE-126, RE-127, RE-128, and RE-135 during a fuel handling accident (FHA) at Point Beach Nuclear Plant (PBNP). Monitors RE-126, RE-127, and RE-128 are located in the reactor containment building (RCB). Monitor RE-135 is inside the Auxiliary (AUX) Building near the Spent Fuel Pool (SFP). The accident occurs either in the SFP or RCB 408 hours after shutdown. The results are used as threshold values for Emergency Action Level (EAL) RA2.2 in the PBNP EAL Technical Basis document, which implements NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors". The containment building, the auxiliary building, and components within the buildings are modeled simplistically because only order of magnitude results are needed. As such, the dose rate results should be considered as reasonably representative of the magnitude of the actual dose rate only. This calculation is not Nuclear Safety Related as the results of the calculation does not affect the design basis or Safety Related systems structures or components. This calculation represents an as built analysis of plant conditions, therefore no acceptance criteria is required.

2. Summary of Results and Conclusion

The results of this calculation are listed below.

Table 2-1 Detector Response

Location	Monitor	Dose Rate (R/h)
RCB	RE-126	5.73
	RE-127	5.57
	RE-128	5.57
SFP	RE-135	3.84

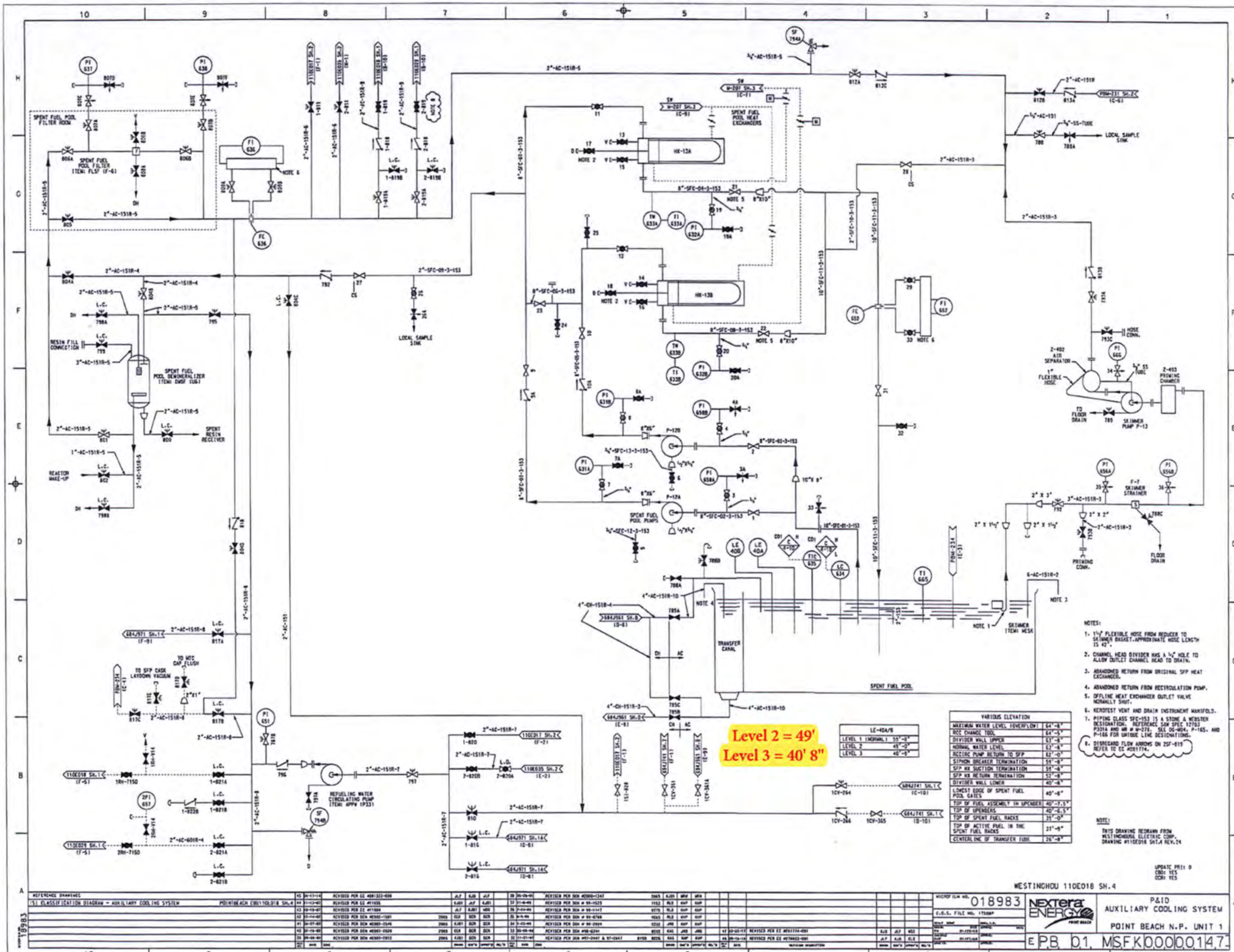
Reading levels at or above the values listed in Table **2-1** will be indicative of a fuel handling accident. The dose rates reported do not include the background (ambient) radiation readings associated with the monitor calibration.

For monitors 126, 127, and 128, a normal background level of 1.5 R/hr was added to the calculated value, and the resulting values were then rounded to whole numbers as follows for use in EAL RA2.2.

RE-126 = 7 R/hr
RE-127 = 7 R/hr
RE-128 = 7 R/hr

SFP monitor RE-135 does not have an existing background reading of significance, so the calculated value was rounded as follows for use in EAL RA2.2:

RE-135 = 4 R/hr



Level 2 = 49'
Level 3 = 40' 8"

SPENT FUEL POOL

LE-404/B	
LEVEL 1 (NORMAL)	59'-0"
LEVEL 2	49'-0"
LEVEL 3	40'-8"

VARIOUS ELEVATION	
MAXIMUM WATER LEVEL (OVERFLOW)	64'-8"
RCC CHANGE TOOL	64'-5"
DIVIDER WALL UPPER	63'-8"
NORMAL WATER LEVEL	62'-8"
RECIRC PUMP RETURN TO SFP	62'-0"
SIPHON BREAKER TERMINATION	59'-8"
SFP HX SUCTION TERMINATION	55'-8"
SFP HX RETURN TERMINATION	52'-8"
DIVIDER WALL LOWER	40'-8"
LOWEST EDGE OF SPENT FUEL POOL GATES	40'-6"
TOP OF FUEL ASSEMBLY IN UPENDER	40'-7.5"
TOP OF UPENDERS	40'-6.5"
TOP OF SPENT FUEL RACKS	35'-0"
TOP OF ACTIVE FUEL IN THE SPENT FUEL RACKS	31'-9"
CENTERLINE OF TRANSFER TUBE	26'-8"

NOTES:

1. 1½" FLEXIBLE HOSE FROM REDUCER TO SKIMMER BASKET. APPROXIMATE HOSE LENGTH IS 42".
2. CHANNEL HEAD DIVIDER HAS A ¼" HOLE TO ALLOW OUTLET CHANNEL HEAD TO DRAIN.
3. ABANDONED RETURN FROM ORIGINAL SFP HEAT EXCHANGER.
4. ABANDONED RETURN FROM RECIRCULATION PUMP.
5. OFFLINE HEAT EXCHANGER OUTLET VALVE NORMALLY SHUT.
6. KEROTEST VENT AND DRAIN INSTRUMENT MANIFOLD.
7. PIPING CLASS SFC-153 IS A STONE & WEBSTER DESIGNATION. REFERENCE S&W SPEC 12703 P301A AND WR # W-278. SEE DG-M04, P-165, AND P-166 FOR UNIQUE LINE DESIGNATIONS.
8. DISREGARD FLOW ARROWS ON 2SF-819 REFER TO EC #281774.

NOTE:

THIS DRAWING REDRAWN FROM WESTINGHOUSE ELECTRIC CORP. DRAWING #110E018 SHT.4 REV.24

UPDATE PRI: 0
CBD: YES
OCR: YES

WESTINGHOU 110E018 SH.4

MICROFILM NO. 018983

C.G.S. FILE NO. 1750BP

NEXTera ENERGY POINT BEACH

P&ID
AUXILIARY COOLING SYSTEM

POINT BEACH N.P. UNIT 1

E P.B. 01. M S F K 0 0 0 0 0 1 4 7

Local Operations for Normal Operation/Shutdown/Cooldown

Step-by-step analysis was performed on the following PBNP procedures to determine those "*rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown*":

- OP 3A: Power operation to hot standby
- OP 3B: Reactor Shutdown
- OP 3C: Hot Standby to Cold Shutdown
- OP 7A: Placing RHR System in operation
- OP 5D Part 4: Degassing the RCS using the PZR and Letdown Gas Stripper

Analysis did not include rooms or areas in which actions of a contingent or emergency nature would be performed. (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations).

Analysis also did not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Procedure:	Procedure Step:	Mode:	Action:	Backup to CR or automatic action?	Vital to plant ops	Location of vital action:
OP 3A	5.2.3	1	Start MFP seal water pumps	Y	N	
OP 3A	5.4.4	1	SW overboard alignment	N	N	
OP 3A	5.5.3	1	shut blast damper	N	N	
OP 3A	5.9.3	1	Open MSR purge valves	N	N	
OP 3A	5.11.1	1	Bypass LP Feed heater coolers	N	N	
OP 3A	5.11.5	1	shut mov-1 and 2	y	N	
OP 3A	5.13.5	1	start turbine bearing lift oil pumps	y	N	
OP 3A	5.20.4	1	lube oil cooldown	n	N	
OP 3B	none	n/a	n/a	n/a	n/a	

Procedure:	Procedure Step:	Mode:	Action:	Backup to CR or automatic action?	Vital to plant ops	Location of vital action:
OP 3C	5.1.1	3/4/5	Put the flex pumps in 8' fan room	N	N	
OP 3C	5.1.7	3/4/5	Sample blender output	N	Y	VCT Area
OP 3C	5.1.8	3/4/5	degas the RCS	N	Y	See OP 5D P4 below
OP 3C	5.6	3/4/5	N2 to the VCT	N	Y	VCT Area
OP 3C	5.12	4/5	Isolate accumulators	N	Y	C-59 area
OP 3C	5.13	4	align flex accumulator	N	N	
OP 3C	5.16	4	Isolate SI pump	Y	N	backup actions to CR only
OP 3C	5.23.6	5	Align containment purge	N	N	
OP 3C	5.23.7	4	align flex N2	N	N	Restraint for entry into Mode 5 only, plant is stable and can remain in this condition until area accessible
OP 3C	attachment A	3/4/5	borate to refueling concentration	Y	N	backup to CR only
Stop at step 5.24 because plant is now in mode 5.						

Procedure:	Procedure Step:	Mode:	Action:	Backup to CR or automatic action?	Vital to plant ops	Location of vital action:
OP 7A	5.1.3a	4	adjust CC cooling to RHR	N	Y	Pipeway 2/3, 8' elev.
OP 7A	5.1.4	4	caution tag 851 871s	N	Y	C-59 area
OP 7A	5.1.5	4	align RHR suction	N	Y	C-59 area, -5ft
OP 7A	5.1.6	4	shut RH-716	N	N	
OP 7A	5.2.16	4, 5	CCW temp control	N	Y	CCW HX Room
OP 5D P4	5.2	3	Initiate primary degas	N	Y	primary sample room
OP 5D P4	5.3.3	3	tag shut CV-261c	N	Y	VCT area
OP 5D P4	5.3.5	3	isolate H2 to vct	N	Y	VCT area

Resulting Tables used in EALs RA3 and HA5 are shown below:

Table R-2 SAFE OPS, S/D, C/D AREAS	
Area/Building	**MODE**
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B-32 MCC Area	4

Table H-2 SAFE OPS, S/D, C/D AREAS	
Area/Building	**MODE**
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B32 MCC Area	4

6.5 LEAKAGE DETECTION SYSTEMS

The leak detection systems reveal the presence of significant leakage from the reactor coolant, residual heat removal, and component cooling systems.

6.5.1 DESIGN BASIS

Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by equipment which permits continuous monitoring of containment air activity and humidity, and of runoff from the air recirculation units and containment floor drains to containment Sump A. This equipment provides indication of normal background which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, gaseous activity, humidity, condensate and floor drain runoff and, in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump. See Section 15.4.3 for additional information regarding leak detection requirements for leak-before-break analyses.

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

The following are monitored for radioactivity concentrations during normal operation, anticipated transients, and accident conditions: the containment atmosphere, the exhausts from the 54 in. auxiliary and service building vent, the 46 in. drumming area vent, the 36 in. containment area vents, the 4 in. combined air ejector exhaust vent, the service water discharge from the containment fan coolers, the component cooling loop liquid, the liquid phase of the secondary side of the steam generator, waste disposal system liquid discharge, spent fuel pool heat exchanger service water return, waste distillate discharge, gas stripper building ventilation exhaust, service water discharge, wastewater effluent, steam line atmospheric release, and the condenser air ejector. GDC 17 is also addressed in Section 11.4, Radiation Protection. A continuing environmental monitoring program, discussed in Section 2.0, is maintained.

Principles of Design

The principles for design of the leakage detection systems can be summarized as follows:

1. Increased leakage could occur as the result of failure of pump seals, valve packing glands, flange gaskets, or instrument connections. The maximum leakage rate calculated for these types of failures is 50 gpm which would be the anticipated flow rate of water through the pump seal if the entire seal were wiped out and the area between the shaft and housing were completely open.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the associated unit's 4.16 kV Class 1E safeguards buses, A05 and A06, utilizing the associated unit's 345/13.8 kV (X03) transformer or the opposite unit's 345/13.8 kV (X03) transformer with the gas turbine in operation, and the associated unit's 13.8/4.16 kV (X04) transformer;

b. One circuit between the offsite transmission network and the opposite unit's 4.16 kV Class 1E safeguards buses, A05 and A06; and

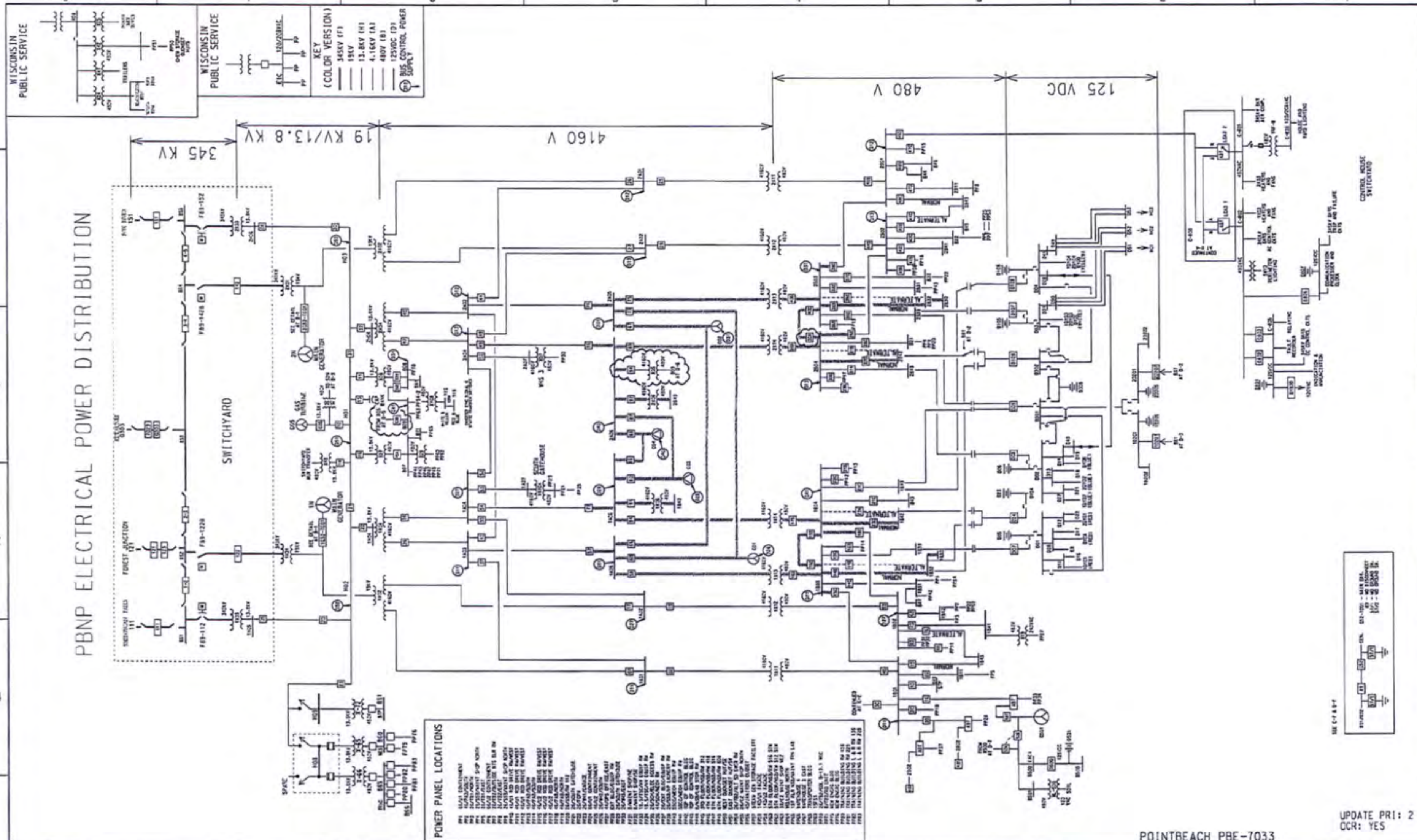
c. One standby emergency power source capable of supplying each 4.16 kV/480 V Class 1E safeguards bus.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to standby emergency power sources.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Associated unit 345/13.8 kV (X03) transformer inoperable. <u>OR</u> Gas turbine not in operation when utilizing opposite unit's 345/13.8 kV (X03) transformer.	A.1 Verify one circuit between the offsite transmission network and the associated unit's 4.16 kV Class 1E safeguards buses, A05 and A06, utilizing the opposite unit's 345/13.8 kV (X03) transformer. <u>AND</u>	24 hours (continued)



UPDATE PRI: 2
OCR: YES

POINTBEACH PBE-7033

REFERENCE DRAWINGS	
SINGLE LINE DIAGRAM	BECH 6118 E-6 SH.1,2,3,4

REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPR'D
11	03-25-17	REVISED PER EC #271218-016 & #279308-007	JLF	KJB	NSS
10	03-21-16	REVISED PER EC #284788-006	JLF	KJB	JSH
09	03-22-16	REVISED PER EC #271218-007	JLF	KJB	MGF
08	10-25-15	REVISED PER EC #295007-000	KJB	JLF	KJB
07	05-12-15	REVISED PER EC #282273-000	KJB	JLF	KJB
06	10-07-14	REVISED PER EC #280675-000	JLF	KJB	GWS
05	04-03-13	REVISED PER EC #278818	JLF	KJB	JLF
04	03-06-13	REVISED PER EC #278362	KJB	JLF	KJB
03	03-01-12	REVISED PER EC #258507	JPH	KJB	JS
02	02-06-12	REVISED PER EC #274521	GJC	JLF	CLF
01	04-06-11	REVISED AND REFORMATTED PER EC #15285	KJB	JLF	KJB
00	07-10-10	INCORP. INTO PDS PER EC #15150	KJB	JLF	KJB

DRAWN DATE
CHECKED DATE
APPROVED DATE
SCALE NONE
FILE NO. PB07322

NEXTera ENERGY POINT BEACH

SIMPLIFIED ELECTRICAL POWER DISTRIBUTION DIAGRAM

POINT BEACH N.P. UNIT 1 & 2

MICROFILM NO. PB07322 | REVISION NO. 11 | DRAWING SIZE D

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—Operating

LCO 3.8.4 The D-01, D-02, D-03, and D-04 DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DC electrical power subsystem inoperable.	-----------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems—Operating," when any DC bus is de-energized. --		
	A.1	Restore DC electrical power subsystem to OPERABLE status.	2 hours
B. Required Action and Associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

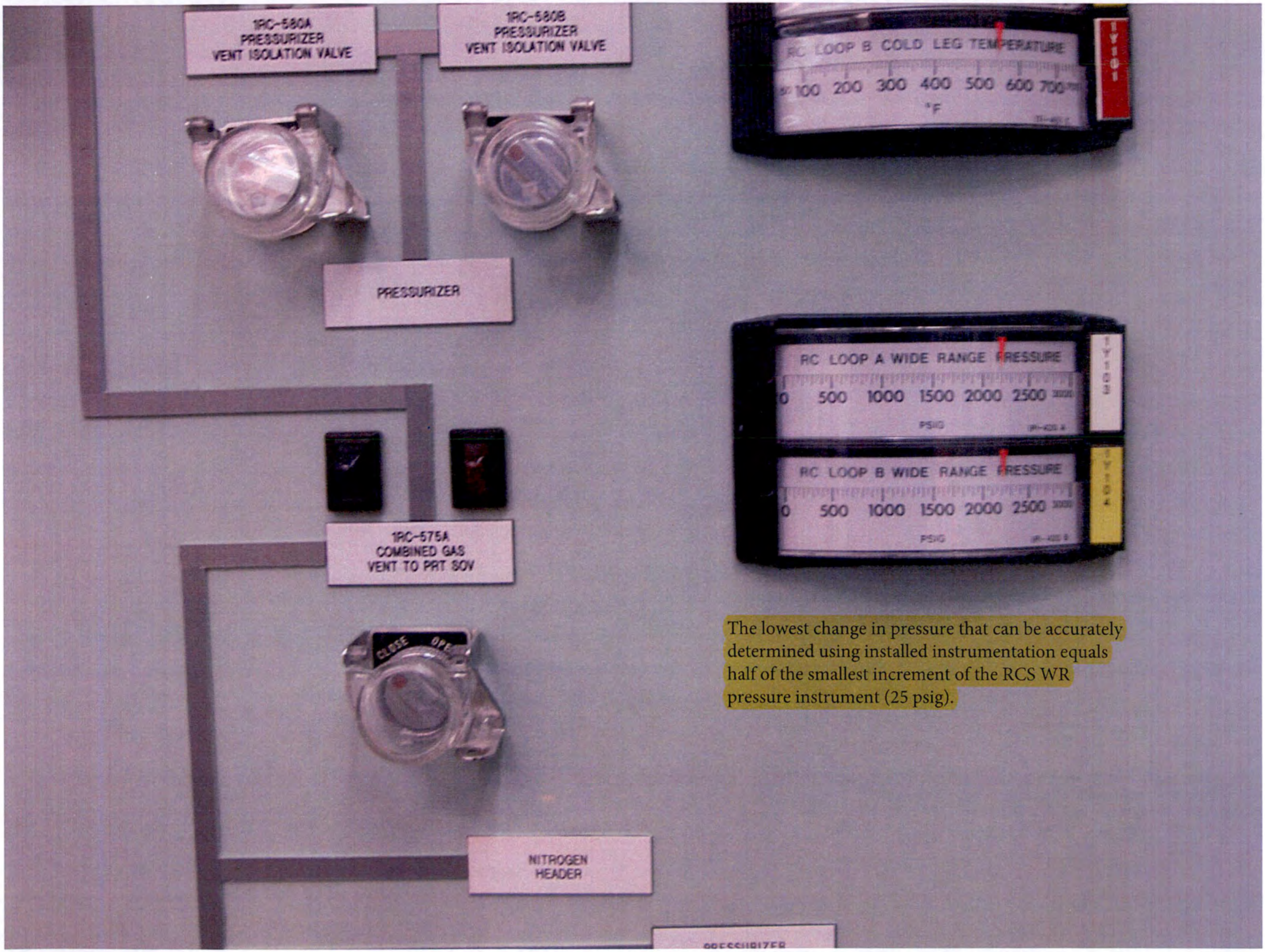
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify correct battery terminal voltage is within limits on float charge.	In accordance with the Surveillance Frequency Control Program

(continued)

EMERGENCY OPERATING PROCEDURE (EOP) SETPOINTS

PNBP STPT	ERG STPT	PARAMETER	DESCRIPTION	SETPOINT Unit 1	SETPOINT Unit 2	CALC REFERENCE
F.18	Y.01	Rx Vessel Level (shutdown)	Minimum shutdown reactor vessel level for operation of RHR without air binding the suction	16%	16%	2010-0027
F.19		Rx Vessel Level (shutdown)	Shutdown reactor vessel level range for indication that level is on scale	10% to 90%	10% to 90%	EOPSTPT Doc F.19
F.20	L.07	Rx Vessel Level	Wide range level corresponding to a void fraction of 25% with both RCPs running, including normal channel accuracy	65 ft	65 ft	2010-0018
F.21			SPARE			2010-0018
F.22	L.11	Rx Vessel Level	Wide Range reactor vessel level corresponding to a void fraction of 0% with both RCPs running	85 ft	85 ft	2010-0018
F.23	L.12	Rx Vessel Level	Wide Range reactor vessel level corresponding to a void fraction of 0% with one RCP running	45 ft	45 ft	2010-0018
F.24	Y.02	Rx Vessel Level	Shutdown reactor vessel level corresponding to mid-loop elevation in the RCS hot legs, including allowances for normal channel accuracies.	20%	20%	2010-0027
F.25	Y.03	Rx Vessel Level	Shutdown reactor vessel level corresponding to just onscale in the RCS hot legs, including allowances for normal channel accuracies.	4%	4%	2010-0027
F.26	Y.04	Rx Vessel Level	Shutdown reactor vessel level corresponding to the top of the RCS hot legs, including allowances for normal channel accuracies.	33%	33%	2010-0027
G.1	M.01	SG Level (narrow range)	Steam generator narrow range no-load level	64%	64%	2010-0019
G.2	M.02	SG Level (narrow range)	Steam generator narrow range level indicating U-tubes covered, including normal channel accuracy and process errors	32%	32%	2010-0019



The lowest change in pressure that can be accurately determined using installed instrumentation equals half of the smallest increment of the RCS WR pressure instrument (25 psig).

Lake Level

The nominal water level in Lake Michigan at the time of the original license submittal was -2 feet relative to the Plant Datum. A maximum water level was recorded in 1886 at +1.7 feet and minimum recorded to date occurred in 1964 at -4.8 feet. The site is, on average, about 20 or more feet above plant elevation zero and there is no record that it has been flooded by the lake.

The maximum analyzed value for high lake level is +1.7 feet. Operators will take actions to commence the orderly shutdown of any operating reactor per Abnormal Operating Procedure direction prior to reaching the analyzed limit.

Flood Level

The license basis level for protection of critical equipment from lake flooding is +9.0 feet (Reference 25). This is an acceptable and bounding value as each lake flooding source when evaluated individually, or in the combined effects review, provides resultant flood levels conservatively below this threshold thereby satisfying the General Design Criteria 2 requirement to include "an appropriate margin for withstanding forces greater than recorded to reflect uncertainties about the historical data and their suitability as a basis for design."

The limiting design basis lake flood event is a combination of the maximum lake level, the maximum wave run-up and a conservative value for the wind setup effect. Details are provided in the following section entitled "Combined Effects." The calculated level reaches +7.25 feet on the riprap shoreline and +8.42 feet on the vertical surfaces of the intake structure (Reference 28). All critical plant components are therefore protected by the strategies outlined in Appendix A.7 "Plant Flooding,"(Reference 30) .

Tides

Tides on Lake Michigan created by the attraction of the moon and sun are insignificant. The total range of oscillation does not exceed 2 inches.

Surges

Using the method delineated in "The Prediction of Surges in the Southern Basin of Lake Michigan, Part I, The Dynamical Basis for Prediction" by G. W. Paltzman (Reference 31), the storm surge that could occur at the site will be 4.14 feet due to the passage of a squall line with a pressure jump of 8 millibars and a simultaneous speed of movement of 65 knots with a shoaling factor of 3.5. Adding this surge of 4.14 feet to the maximum recorded water level in Lake Michigan of +1.7 feet results in a maximum elevation of 5.84 feet, which is bounded by the license basis flood level and is considerably lower than the turbine building grade floor elevation of +8.0 feet or the pumphouse operating floor elevation of +7.0 feet.

The value of 4.14 feet was developed using Platzman's contours of amplitude for pressure. There are no contours for the lake in the area of the site so a conservative approach was taken using the reflected surge values for Waukegan at 90° with a speed of movement of 65 knots, giving a pressure rise of 0.05 feet. Using 8 millibars or 0.236", the maximum surge due to pressure with a 3.5 shoaling factor will, therefore, be

3.7.7 Service Water (SW) System

TLCO 3.7.7 The SW System shall be operated in accordance with the evaluated conditions and alignments of an analyses demonstrating the ability to provide required cooling water flow to required equipment.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

-----NOTE-----
Performance of an evaluation, within the required Completion Time, that verifies the SW System is capable of providing required cooling water flow to required equipment is an acceptable alternative to the stated Required Actions.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SW Intake temperature > 85°F.	A.1	Declare G-01, G-02, Containment Fan Coolers, PAB Battery Room Vent Coolers, Component Cooling Water (CCW) heat exchangers (HX), inoperable.	Immediately
B. Pump Bay level < -11.5 feet.	B.1	Declare Unit 1 "A", Unit 2 "A", and Unit 2 "B" accident fan cooler units inoperable.	Immediately
C. Pump Bay level < -15 feet	C.1	Declare G01 and G02 inoperable and all containment fan coolers inoperable.	Immediately

(continued)

RESULTS AND CONCLUSIONS

The SW pumps will perform adequately with a minimum pumpbay water level at the –19 (ft) elevation and an individual pump flowrate up to 7800 (gpm). A water level below the –19 (ft) elevation could result in vortexing due to inadequate inlet submergence. Cavitation should not occur since adequate NPSH will be available down to the –26 (ft) elevation. Since the pump inlet is located at the –26 (ft) elevation, the pump would become airbound prior to cavitation.

Pumpbay water level is far superior to pump discharge pressure for indication of the onset of inadequate pump performance. Adequate pump performance is a function of both submergence and flowrate inputs (flowrate is used directly in Equation (7) and indirectly in Equation (6) to determine NPSHR). ***Discharge pressure alone cannot serve as an indicator of cavitation because the fluctuating pumpbay level is not accounted for.*** Since minimum submergence has been determined for a conservatively high flowrate, pumpbay level should be used in lieu of discharge pressure to directly indicate the onset in inadequate pump performance.

The following documents and actions have been initiated by this calculation.

- Engineering Work Request (EWR)– Since the discharge pressure setpoint is not based on cavitation prevention, an EWR will be initiated to evaluate the proper source for the setpoint.
- Potential Open Item For DBD Program (PBF-1611) – Service Water Design Basis Document
- Procedure Feedback Request (PBF-0026p) – OI-70, Rev. 31, Sec. 3.13 will be revised to reflect the independence of cavitation prevention from discharge pressure setpoint.
- Training Department Notification – Training Reference Handbook (TRHB) 11.8, Rev.7, Sec. 5.2.3 should be revised to reflect the independence of cavitation prevention from discharge pressure.

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

EPFAQ Number:	2016-002
Originator:	David Young
Organization:	NEI
Relevant Guidance:	NEI 99-01, *Methodology for Development of Emergency Action Levels*, Revisions 4 and 5; and NEI 99-01, *Development of Emergency Action Levels for Non-Passive Reactors*, Revision 6. NUMARC/NESP-007, *Methodology for Development of Emergency Action Levels.*
Applicable Section(s):	Initiating Condition (IC) HA2 in NEI 99-01, Revisions 4 and 5, and NUMARC/NESP-007, "FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown" ICs CA6 and SA9 in NEI 99-01, Revision 6: "Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode" Definition of VISIBLE DAMAGE in NEI 99-01, Revisions 4, 5 and 6, and NUMARC/NESP-007
Status:	Available for Public Comment

NOTE:

Based on industry comments provided by letter dated February 16, 2017 (ADAMS Accession No. ML17079A228), and subsequent staff discussions, a proposed revision to these ICs was proposed in the public meeting held on April 4, 2017, and was attached to the public meeting notice (ADAMS Accession No. ML17089A458). Based on comments provided by the industry during the April 4, 2017 public meeting, the staff revised the proposed revisions to these ICs.

QUESTION OR COMMENT:

A review of industry Operating Experience has identified a need to clarify an aspect of the definition of VISIBLE DAMAGE as it relates to the ICs cited above; adding this clarity is necessary to minimize the potential for an over-classification of an equipment failure. There may be cases where VISIBLE DAMAGE is the result of an equipment failure and limited to the failed component (i.e., the failure did not cause damage to any other component or a structure). The current definition of VISIBLE DAMAGE does not adequately differentiate between damage resulting from, and affecting only, the failed piece of equipment vs. an equipment failure causing damage to another component or a structure (e.g., by a failure-induced fire or explosion). Can the definition of VISIBLE DAMAGE be clarified to help avoid an inappropriate emergency declaration in cases where an equipment failure does not result in damage to another component or a structure (i.e., VISIBLE DAMAGE affects only the failed component)?

A related question is also posed – Consistent with the approach used in other ICs, should a note be added to preclude an emergency declaration if the safety system affected by a hazard was not functional before the event occurred (e.g., tagged out for maintenance)?

PROPOSED SOLUTION:

Yes; the sentence below may be added to the definition of VISIBLE DAMAGE [as defined in NEI 99-01, Revisions 4, 5, and 6].

Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

From a plant safety and change-in-risk perspective, the consequences from the failure of a piece of equipment, accompanied by a hazard (e.g., a fire or explosion) that does not damage

any other equipment or a structure, are essentially the same as the equipment failing with no attendant hazard. Neither event would appear to meet the definition of an Alert because the outcome does not involve an actual or potential substantial degradation of the level of safety of the plant (e.g., there has been no significant reduction in the margin to a loss or potential loss of a fission product barrier). Nuclear power plants are designed with redundant safety system trains that are required to be separated (i.e., installed in separate plant areas or have separation within an individual area).

Absent any collateral damage to another component or a structure, a hazard associated with an equipment failure does not affect the ability to protect public health and safety, and there is no additional response benefit to be gained by declaring an emergency. The normal plant organization has sufficient resources and adequate guidance to respond to an equipment failure – guidance includes operating procedures and Technical Specifications; the fire protection [program], industrial safety and corrective action programs; and work management and maintenance requirements.

Concerning the second question, an emergency declaration would not be appropriate in response to a hazard affecting a piece of equipment or system that was non-functional prior to the event (e.g., tagged out for maintenance). For this reason and consistent with the approach used in other ICs, the following note may be added to IC HA2 (NEI 99-01 R4 and R5), or ICs CA6 and SA9 (NEI 99-01 R6).

Note: If the affected safety system (or component) was already non-functional before the event occurred, then no emergency classification is warranted.

Consistent with the guidance in Regulatory Issue Summary (RIS) 2003-18, Supplement 2, *Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4*, dated January 2003, it is reasonable to conclude that the changes proposed above would be considered as a "deviation."

NRC RESPONSE:

The proposed guidance is intended to ensure that an Alert should be declared only when actual or potential performance issues with SAFETY SYSTEMS have occurred as a result of a hazardous event. The occurrence of a hazardous event will result in a Notification of Unusual Event (NOUE) classification at a minimum. In order to warrant escalation to the Alert classification, the hazardous event should cause indications of degraded performance to one train of a SAFETY SYSTEM with either indications of degraded performance on the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second SAFETY SYSTEM train, such that the operability or reliability of the second train is a concern. In addition, escalation to the Alert classification should not occur if the damage from the hazardous event is limited to a SAFETY SYSTEM that was inoperable, or out of service, prior to the event occurring. As such, the proposed guidance will reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, i.e., does not cause significant concern with shutting down or cooling down the plant.

IC HA2 (NEI 99-01 R4 and R5; NUMARC/NESP-007), or ICs CA6 and SA9 (NEI 99-01 R6), do not directly escalate to a Site Area Emergency or a General Emergency due to a hazardous event. The Fission Product Barrier and/or Abnormal Radiation Levels/Radiological Effluent recognition categories would provide an escalation path to a Site Area Emergency or a General Emergency.

The proposed addition of the following notes, applicable to ICs HA2 (NEI 99-01 R4 and R5; NUMARC/NESP-007), or ICs CA6 and SA9 (NEI 99-01 R6), provide further clarification as to how these Alert emergency classifications are considered. The revisions to these EALs, including the addition of the notes, are consistent with the current NRC-endorsed Alert

classification language.

1. Adding the following note to the applicable EALs, per this EPFAQ, is acceptable as it meets the intent of the EALs, is consistent with other EALs (e.g., EAL HA5 from NEI 99-01, Revision 6; this revision was endorsed by the NRC in a letter dated March 28, 2013, available at ADAMS Accession No. ML12346A463), and ensures that declared emergencies are based upon unplanned events with the potential to pose a radiological risk to the public.

   If the affected safety system train (or component) was already inoperable or out of service before the event occurred, then this emergency classification is not warranted as long as the damage is limited to the affected safety system train (or component).

2. Adding the following note to help explain the EAL is reasonable to succinctly capture the more detailed information from the Basis section related to when conditions would require the declaration of an Alert.

   If the event results in VISIBLE DAMAGE, with no indications of degraded performance to any SAFETY SYSTEM train, then this emergency declaration is not warranted.

Revising the EALs and the Basis sections to ensure potential escalations from a NOUE to an Alert, due to a hazardous event, is appropriate as the concern with these EALs is: (1) a hazardous event has occurred, (2) one SAFETY SYSTEM train is having performance issues as a result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or the VISIBLE DAMAGE is enough to be concerned that the second SAFETY SYSTEM train may have operability or reliability issues.

Revising the definition for VISIBLE DAMAGE is appropriate as this definition is only used for these EALs and the revised EALs are based upon SAFETY SYSTEM trains rather than individual components or structures.

All of the changes discussed above are addressed in the attached markups to NEI 99-01, Revision 6. Licensees that use NESP-007, NEI 99-01 Revision 4, or NEI 99-01 Revision 5 EAL schemes can adopt this language in the relevant format the staff approved for their use.

Consistent with the guidance in Regulatory Issue Summary (RIS) 2003-18, Supplement 2, *Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4*, dated January 2003, it is reasonable to conclude that the changes proposed (discussed above and as attached) would be considered as a "deviation."

RECOMMENDED FUTURE ACTION(S):

☐ INFORMATION ONLY, MAINTAIN EPFAQ

☒ UPDATE GUIDANCE DURING NEXT REVISION

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

CA6

ECL: Alert

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

(1) a. The occurrence of **ANY** of the following hazardous events:
- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- (site-specific hazards)
- Other events with similar hazard characteristics as determined by the Shift Manager

AND

b. 1. Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:
- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
- Event damage has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE

such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria 1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC AS1.

Developer Notes:

For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).

Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.

ECL Assignment Attributes: 3.1.2.B

SA9

ECL: Alert

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

(1) a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- (site-specific hazards)
- Other events with similar hazard characteristics as determined by the Shift Manager

AND

b. 1. Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
- Event damage has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE

such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria 1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via ICs FS1 or AS1.

Developer Notes:

For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).

Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.

ECL Assignment Attributes: 3.1.2.B

VISIBLE DAMAGE: Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM train.

1. PURPOSE AND SCOPE

The purpose of this calculation is to evaluate dose rates as a function of water height in the reactor vessel during cold shutdown or refueling operations in order to set Emergency Action Level (EAL) thresholds for core uncovery (RA2, CS1, CG1). The dose rates are calculated at the locations of the containment monitors RE-126, RE-127 and RE-128 so that dose rate measurements by these devices can be correlated to the water level in the core, upon failure of other water level detection systems. This calculation will determine the dose rate at full core uncovery, as well as maximum water levels with a detectable dose rate response applicable to both Unit 1 and Unit 2. This calculation is not Nuclear Safety Related as the results of the calculation do not affect the design basis or Safety Related systems structures or components. These results are best estimates based on as-built conditions and provide information to operators with respect to classifying an emergency, therefore no acceptance criteria is required.

2. SUMMARY OF RESULTS AND CONCLUSION

The dose rate results for the configuration without the reactor vessel head and with the reactor vessel head are provided in Section 7.8.1 and Section 7.8.2, respectively. The minimum dose rates with the core uncovered (i.e. water at the top of the active fuel) are shown in the table below. The dose rates reported below do not include the ambient readings associated with the monitor calibration (generally 1 to 2 R/h).

Table 2-1 Dose Rate at Top of Active Fuel

Model Description	Dose Rate (R/h)
Head Off	1.09E+02
Head On	2.94E-02[1]

Detailed results of the dose rate as a function of water height are provided in Table 7-3 for cases with the head removed.

Calculated value for RX head in place is below usable scale for the available radiation monitors and is therefore not used.

Calculated value for RX head in place (109 R/hr) was rounded to 100 R/hr for ease of use in the EALs as an approximate indication of the underlying plant condition being assessed.

[1] For the case with the head in place, the dose rate is below the detectable range of the radiation monitors of 1 R/h.

EOP STEP: 23 ERG STEP: 22

STEP:

Check Containment Hydrogen Concentration:

PURPOSE:

To check if an excessive containment hydrogen concentration is present.

BASIS:

This step instructs the operator to obtain a current hydrogen concentration measurement. Depending upon the magnitude of the hydrogen concentration, the operator will either continue with this procedure or notify the TSC to determine additional recovery actions concerning Post Accident Containment Hydrogen Reduction before continuing with the procedure.

When inadequate core cooling has occurred, the containment hydrogen concentration may be as much as 10 to 12 volume percent, depending on the amount of metal-water reaction (to produce hydrogen) that has occurred in the core. The hydrogen concentration is of concern since a flammable mixture can burn, if an ignition source is available, and cause a sudden rise in containment pressure which may challenge containment integrity. The operator is instructed to obtain a current containment hydrogen concentration measurement at this point in order to ascertain the potential flammability of the combustible gases in the containment. Note that in order to have the potential for flammable hydrogen concentrations, an inadequate core cooling situation must have already existed. Without an inadequate core cooling situation, sufficient hydrogen would not be expected to have been produced to cause potentially flammable mixtures.

A determination is made of the flammability of the hydrogen mixture with respect to the possible containment pressure rise. If the hydrogen mixture is between 0.5 volume percent and 6.0 volume percent in dry air, either no hydrogen burn is possible or a limited burn may occur which does not produce a significant pressure rise. If greater than 0.5 volume percent, the operator is instructed to consult with TSC staff to determine if Post Accident Containment Hydrogen Reduction should be performed to slowly reduce containment hydrogen concentration using ventilation. If the hydrogen concentration is less than 0.5 volume percent in dry air, a flammable situation is not imminent and the operator continues with this procedure.

5.4 HSM or HSM-H Dose Rate Evaluation Program

5.4.1 The licensee shall establish a set of HSM dose rate limits which are to be applied to DSCs used at the site to ensure the limits of 10 CFR Part 20 and 10 CFR 72.104 are met. Limits shall establish peak dose rates at the following three locations:

1) HSM front bird screen,
2) HSM front surface, and
3) Outside HSM door at the HSM door centerline.

5.4.2 Notwithstanding the limits established in 5.4.1, the dose rate limits listed below for the Standardized HSM and HSM-H shall be met when a specific DSC model loaded with fuel is stored within a module:

Dose Rate Limits for the Standardized HSM and HSM-H

DSC Model	HSM Model	Dose Rate Limit HSM Front Bird Screen (mrem/hour)	Dose Rate Limit Outside HSM Door (mrem/hour)	Dose Rate Limit End Shield Wall Exterior (mrem/hour)
24P	Standardized HSM	350	70	55
52B	Standardized HSM	350	70	55
61BT	Standardized HSM	1300	200	15
32PT	Standardized HSM	850	200	6
24PHB	Standardized HSM	525	20	275
24PTH*	Standardized HSM	525	70	300
61BTH	Standardized HSM	200	100	15
24PTH	HSM-H	1300	2	5
61BTH	HSM-H	650	2	4
32PTH1	HSM-H	525	2	2
69BTH	*HSM-H*	*250*	*2*	*4*
37PTH	*HSM-H*	*525*	*2*	*4*

* *Applicable only to 24PTH-S-LC.*

2 times
32 PT DSC limits
(mrem/hr)
HSM Frt = 1,700
HSM door = 400
End Shield Wall Ext. = 12

The number and locations of the dose rate measurements on the surface of front bird screen of the HSM are indicated below:

- Two dose rate measurements are taken for each front bird screen for the HSM-H. These dose rate measurements are approximately within 24 inches measured from the surface of the ISFSI pad and are approximately 6 inches from the centerline of each front bird screen.
- For the standardized HSM models, three dose rates are taken on the surface of each front bird screen. The central dose location shall be at the approximate centerline of the front bird screen. The other two dose locations are spaced at approximately equal distance on either side of the central dose location. All dose locations shall be at least 24 inches above the pad surface.
- None of these measurements shall exceed the specified dose rate limits.

(continued)

1.2.4 Maximum External Surface Dose Rate

Limit/Specification:

The external surface average dose rate from all types of radiation will be less than 100 mrem per hour on the sides and 200 mrem/hr on the top. Dose rates at the air inlets and outlets will be below 350 and 100 mrem/hr, respectively.

Applicability: This dose rate limit shall apply to the entire external surface of the VCC, except the bottom surface.

Objective: The external dose rate is limited to this value to ensure that the cask has not been inadvertently loaded with fuel not meeting the specifications in Section 2.0 of the FSAR, to provide verification for plant personnel that radiation levels are acceptably low, and to satisfy the 10CFR72 dose rate limit of 25 mrem/year at the ISFSI controlled area boundary.

Action: If the measured dose rates are above those values listed above, correct fuel loading shall be verified. If correct fuel is loaded, specific analyses must demonstrate compliance with 10 CFR Part 20 and 10 CFR Part 72 radiation protection requirements, or appropriate action must be taken to comply with the acceptable limits. A letter report, summarizing the action taken and the results of investigation conducted to determine the cause of the high dose rates, shall be submitted to the NRC within 30 days. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to the administrator of the appropriate NRC regional office.

Surveillance: The external surface dose rate shall be measured after loading the MSB in the VCC and before transfer to the storage pad. The side dose rate shall be measured at a distance of 5 feet from the bottom of the VCC and at four equally spaced radial locations. The top dose rate shall be measured at the VCC lid center and the VCC outer lid edge. The dose rate measurement shall account for the effects of background radiation on the absolute dose rate measurements.

2 times VSC-24 limits
sides = 200
top = 400
inlets = 700
outlets = 200

FIGURE 2
ST-2 CORE COOLING

Note: Blockage of the core inlet due to debris passing through the sump screen may occur when on containment sump recirculation. Significant core blockage can be indicated by Core Exit Thermocouples (CETs) indicating greater than saturation when RVLIS shows the core to be covered. If this occurs, RVLIS indications of water level in the fuel are not accurate. Procedure transitions should be based on CET trends (i.e. if CETs are trending up, then the higher level CSP is entered; if CETs are trending down, then the lower level CSP is entered.

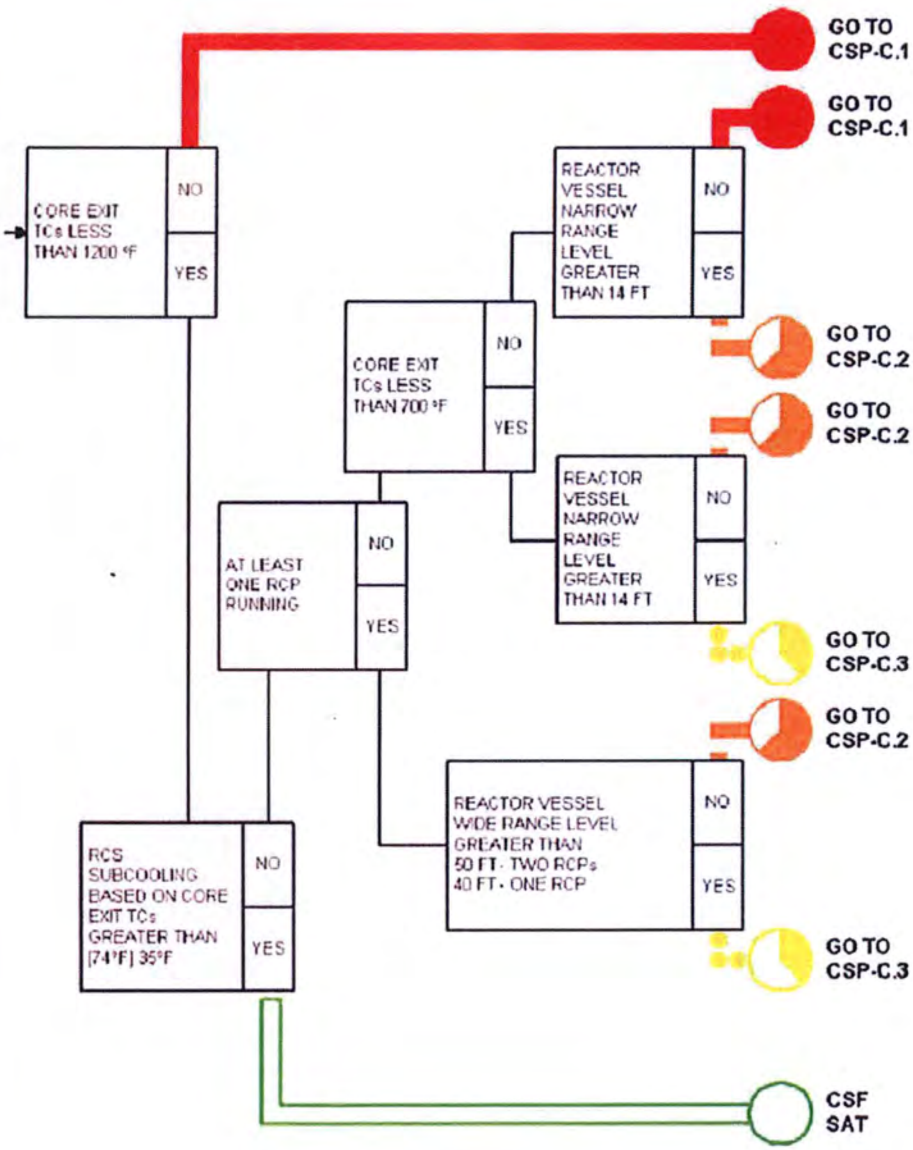
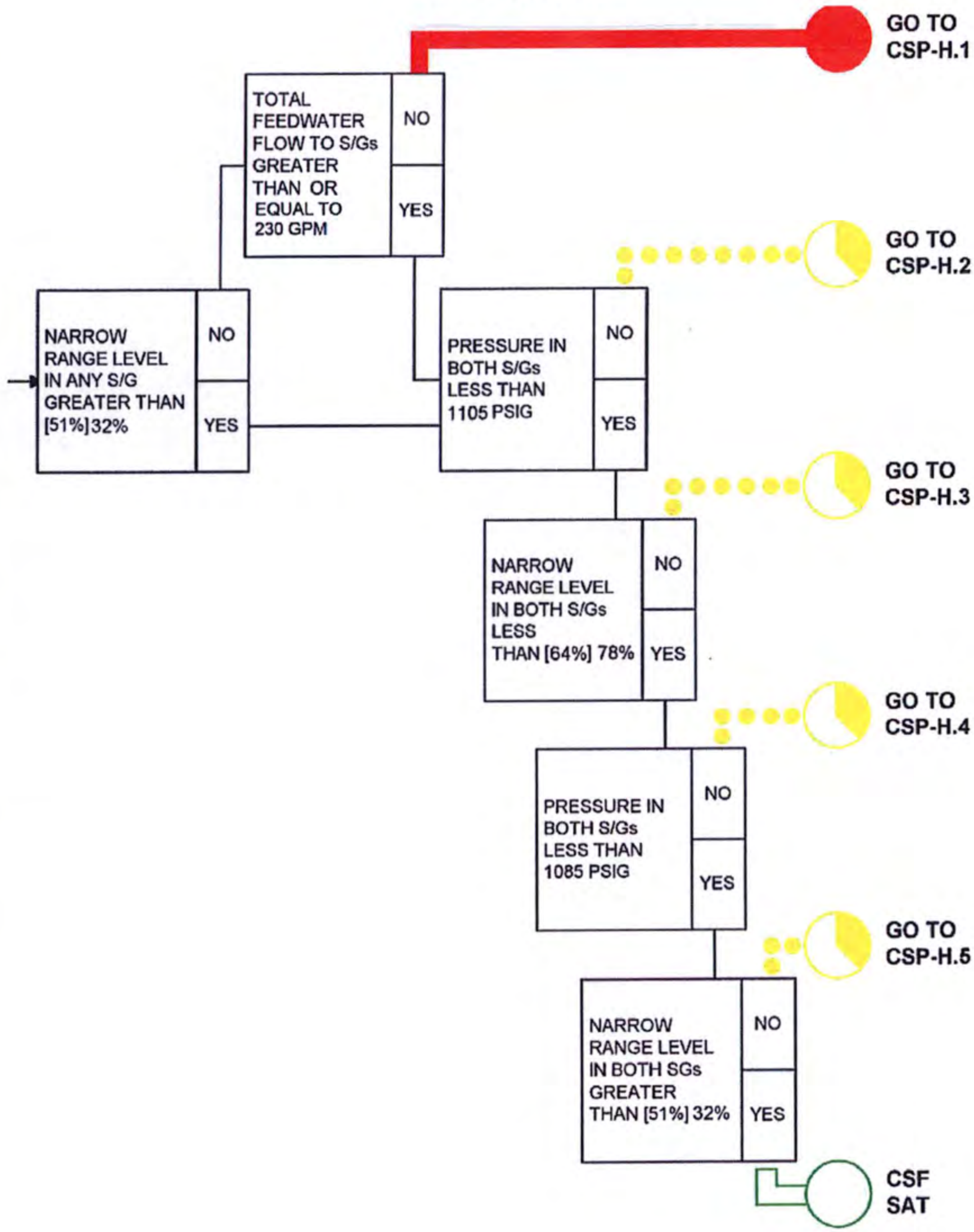


FIGURE 3
ST-3 HEAT SINK



10.0 Results and Conclusions

10.1 Dose Rates-RCS Activity

The initial dose rates (R/hour) are summarized below. The RE-127 dose rates are plotted in Figure 12-1.

RCS Activity: 300μCi/gm DE I-131, 1% fuel defects for non-iodine nuclides

1/2RE-126	814.7
1/2RE-127	577.1
1/2RE-128	645.0
1RE-102	319.1
2RE-102	553.3

577 used based on Opeations feedback

The dose rates have increased to about 34 times the Revision 0 values.

RCS Activity: 1% fuel defects for all nuclides

1/2RE-126	2.99E+01
1/2RE-127	2.12E+01
1/2RE-128	2.37E+01
1RE-102	1.17E+01
2RE-102	2.03E+01

The dose rates have increased to about 1.6 times the Revision 0 values.

10.2 Dose Rates-20% Failed Fuel

1/2RE-126	2.72E+04
1/2RE-127	1.85E+04
1/2RE-128	2.20E+04

18,500 used based on Operations feedback

RE-127 Dose Rates are plotted in Figure 12-2.

The dose rates have increased by 17% over the pre-EPU values.

10.3 Dose Rates-Technical Specification Activity

1/2RE-126	1.42E+01
1/2RE-127	1.10E+01
1/2RE-128	1.21E+01
1RE-102	4.38E+00
2RE-102	7.67E+00

11 used based on Operations feedback

RE-127 dose rates are plotted in Figure 12-3.

The dose rates have increased by about 3.7 times the Revision 0 values.

10.4 Conclusions

The estimated dose rates, for EPU conditions, have increased over those previously calculated for pre-EPU (Revision 0) conditions.

Comparison of the iodine Curies, provided in Sections 9.1 and 9.2, shows that 300 μCi/gm of DE I-131 (based on AST dose conversion factors) results in significantly greater activity than the 1% defect level. Comparing the estimated dose rates from Sections 9.1 and 9.2 shows that 300 μCi/gm of iodine is a significant contributor to the dose rate.

11.0 Tables

This section is empty.

Results and Conclusions:

The results were calculated using a source term corresponding to a reactor thermal power operating level of 1683 MWth. The results are applicable generally to other thermal power operating levels because core activity and photon intensity are directly proportional to the reactor thermal operating power level.

The failed fuel monitor reading at time equal to 0 that would correspond to 300 μCi/g I-131 DE is approximately 4500 mR/hr and the failed fuel monitor reading at time equal to 0 that would correspond to 50 μCi/g I-131 DE is approximately 750 mR/hr.

The calculated RCS total radionuclide activity and fuel clad failure percentage correlations for use with the failed fuel monitor (RE-109) reading are graphically displayed in Figure 1 below. To determine fuel clad failure percentage, total primary system activity, or total I-131 DE activity, multiply the respective conversion factor by the failed fuel monitor reading.

Example: An event occurs causing fuel failure. The failed fuel monitor reading 12 hours after fuel failure occurs is 2000 mR/hour. The conversion factor for fuel clad failure from Figure 1 is approximately 7E-04 % clad failure per mR/hr. The estimated fuel clad failure is 7E-04 x 2000 or 1.4 %.

These factors are acceptable for use when the sample system is not isolated, i.e., a containment isolation signal has not occurred. These factors can be used after sample system isolation has occurred, however, the values will only reflect the condition of the primary system at the time of isolation.

Figure 1
RE-109 Reading/Damage Relationship

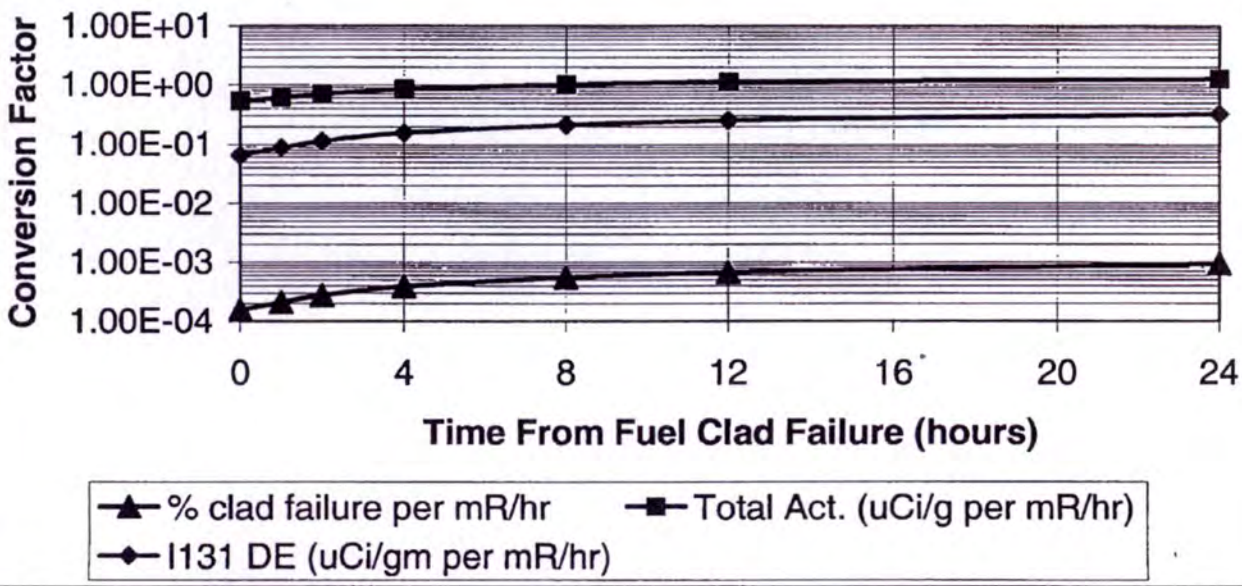
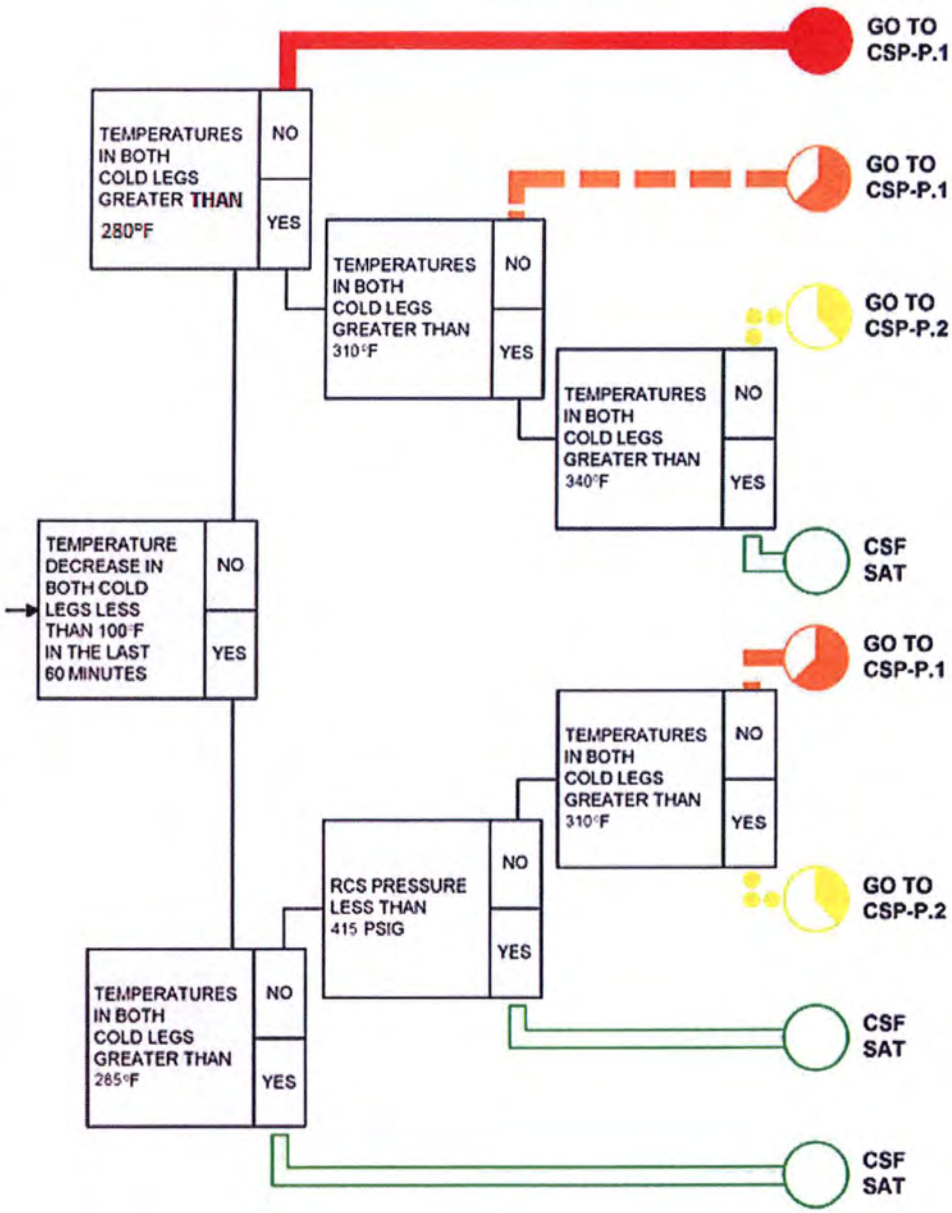


FIGURE 4
ST-4 INTEGRITY



7. Uplift due to buoyant forces

8. External pressure load

The critical loading condition is that caused by the maximum credible accident resulting from severance of a reactor coolant pipe coincident with the maximum hypothetical earthquake.

Loss of Coolant Accident Load

The design pressure and temperature of the containment is in excess of the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical severance of a reactor coolant pipe.

The supports for the reactor coolant system are designed to withstand the blowdown forces associated with the severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible. Transients resulting from the loss of coolant accident and other lesser accidents are presented in Chapter 14 and serve as the basis for a containment design pressure of 60 psig.

The design pressure is not exceeded during any subsequent long term pressure transient caused by the combined effects of such heat sources as residual heat and metal-water reactions. These effects are overcome by the combination of emergency powered engineered safeguards and structural heat sinks.

The temperature gradient through the wall during the loss of coolant accident is shown in Figure 5.1-6. The variation of temperature with time and the expansion of the liner plate are considered in designing for the thermal stresses associated with the loss of coolant accident load.

Structure Dead Load

Dead load consists of the weight of the concrete wall, dome, base slab, and any internal concrete. Weights used for dead load calculations are as follows:

1.	Concrete	143 lb/ft^3
2.	Steel Reinforcing & Prestressing Steel	489 lb/ft^3 using nominal cross sectional areas of reinforcing as defined in ASTM for bar sizes and nominal cross sectional areas of prestressing tendons.
3.	Steel Lining	489 lb/ft^3 using nominal cross sectional area of lining.

EMERGENCY OPERATING PROCEDURE (EOP) SETPOINTS

PNBP STPT	ERG STPT	PARAMETER	DESCRIPTION	SETPOINT Unit 1	SETPOINT Unit 2	CALC REFERENCE
L.26		Flow (Charging)	Maximum charging flow	140 gpm	140 gpm	2010-0022
L.27		Flow (AFW)	Minimum indicated AFW flow to mitigate LONF/LOAC event, including normal uncertainties	315 gpm	315 gpm	2010-0022
L.28	S.09	Flow (RHR)	RHR pump flowrate required to sweep air from the RCS hot leg piping, including allowance for normal channel accuracies.	1550 gpm	1550 gpm	2010-0022
L.29	S.11	Flow (Charging)	Charging flowrate requirement that discriminates between an RCS leak and a small break Loss of Coolant Accident minus the flow rate of the highest capacity letdown orifice, not to exceed maximum VCT makeup capacity.	30 gpm	30 gpm	2010-0022
M.1		Containment (Pressure)	Containment High-High pressure SLI setpoint	15 psig	15 psig	2010-0023
M.2	T.02	Containment (Pressure)	Containment pressure setpoint for spray actuation	25 psig	25 psig	2010-0023
M.3			SPARE			2010-0023
M.4	T.03	Containment (Radiation)	Containment design pressure	60 psig	60 psig	2010-0023
M.5	T.07	Containment (Radiation)	Radiation level alarm setpoint for post-accident containment radiation monitor	10 R/hr	10 R/hr	2010-0023
M.6			SPARE			2010-0023
M.7	T.06	Containment (Sump)	Sump B water level just below design flood level	86 in	86 in	2010-0023
M.8			SPARE			2010-0023

APPENDIX I
SEISMIC
Page 1 of 4

DISCUSSION/OVERVIEW:

Earthquakes are defined as a vibration of the earth's surface that occurs after a release of energy in the earth's crust. Because the earth's crust is made up of numerous segments or "plates" that are constantly moving slowly, vibrations can occur and result in small or large earthquakes. Most earthquakes are quite small and not readily felt. Larger and more violent earthquakes are those that occur in a release of energy as the plates slide past or collide into one another.

PURPOSE:

This document is intended to identify those seismic protection features in-place at the site, and to identify the implementing document(s) that are used to check their function and/or ensure they are capable of continuing to perform their function.

CLB FEATURES:

Horizontal ground acceleration at the site of 0.06g combined with a vertical acceleration of 0.04g are used for the Operating Basis Earthquake (OBE). These accelerations are considered as acting simultaneously. OBE is selected to be typical of the largest probable ground motion based on the site seismic history. Components and systems that are essential for continued operation without undue risk to the health and safety of the public are designed to remain functional due to OBE seismic effects.

The hypothetical earthquake or Safe Shutdown Earthquake (SSE) is selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties. SSE is twice the magnitude of the OBE. The seismic design for the SSE is intended to provide a margin in design that assures the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and maintain it in a safe shutdown condition, the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the exposures of 10 CFR 50.67.

EVENT RESPONSE:

Seismic Detector Annunciator C01 A 2-6 and PPCS Seismic Alarm will go in alarm if 0.01g horizontal or vertical acceleration is recorded. PPCS Seismic Alarm will alarm if 0.06g horizontal acceleration or a 0.04g vertical acceleration is recorded. Either alarm will prompt entry into AOP-28, Seismic Events, as will the event being "felt" by the operations staff, or reported to have occurred by a reliable source. A plant walkdown to determine equipment damage is performed while the data from the seismic recorders is being retrieved and analyzed. If it is determined by the seismic recorder analysis that an Operating Basis Earthquake (OBE) limit has been exceeded, then a focused inspection by Engineering will be performed using PBF-7044, Focused Inspection.

Matrix of Table H-1 Fire Areas below was developed by review of AOP-40A, "Control Room Abandonment Due To Fire" and the PBNP Shutdown Safety Plan. Listing was then verified against the Vital Areas of the PBNP Security Plan to identify any other potential areas containing safety-related equipment.

	AOP-40A	Shutdown Safety Plan	Security Vital Areas
G05 Building	x	x	
13.8kV building	x	x	
Cable Spreading Room	x	x	x
Vital Switchgear Room	x	x	x
AFW Room	x	x	x
G01/G02 Rooms	x	x	x
EDG building	x	x	x
PAB (all elevations)	x	x	x
Service Water Pump Rooms		x	x
Containment		x	x
Control Room	x	x	x
Facade 85'	x		x (by default behind PAB VA door)

The resulting site-specific list of plant rooms or areas for PBNP was captured in Table H-1 below:

Table H-1 Areas
Control Room
Containment
PAB
G05 building
13.8kV Building
Cable Spreading Room
Vital Switchgear Room
AFW Pump Room
G-01/02 Rooms
EDG Building
Service Water Pump Rooms
Façade 85'

Local Operations for Normal Operation/Shutdown/Cooldown

Step-by-step analysis was performed on the following PBNP procedures to determine those "*rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown*":

- OP 3A: Power operation to hot standby
- OP 3B: Reactor Shutdown
- OP 3C: Hot Standby to Cold Shutdown
- OP 7A: Placing RHR System in operation
- OP 5D Part 4: Degassing the RCS using the PZR and Letdown Gas Stripper

Analysis did not include rooms or areas in which actions of a contingent or emergency nature would be performed. (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations).

Analysis also did not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Procedure:	Procedure Step:	Mode:	Action:	Backup to CR or automatic action?	Vital to plant ops	Location of vital action:
OP 3A	5.2.3	1	Start MFP seal water pumps	Y	N	
OP 3A	5.4.4	1	SW overboard alignment	N	N	
OP 3A	5.5.3	1	shut blast damper	N	N	
OP 3A	5.9.3	1	Open MSR purge valves	N	N	
OP 3A	5.11.1	1	Bypass LP Feed heater coolers	N	N	
OP 3A	5.11.5	1	shut mov-1 and 2	y	N	
OP 3A	5.13.5	1	start turbine bearing lift oil pumps	y	N	
OP 3A	5.20.4	1	lube oil cooldown	n	N	
OP 3B	none	n/a	n/a	n/a	n/a	

Procedure:	Procedure Step:	Mode:	Action:	Backup to CR or automatic action?	Vital to plant ops	Location of vital action:
OP 3C	5.1.1	3/4/5	Put the flex pumps in 8' fan room	N	N	
OP 3C	5.1.7	3/4/5	Sample blender output	N	Y	VCT Area
OP 3C	5.1.8	3/4/5	degas the RCS	N	Y	See OP 5D P4 below
OP 3C	5.6	3/4/5	N2 to the VCT	N	Y	VCT Area
OP 3C	5.12	4/5	Isolate accumulators	N	Y	C-59 area
OP 3C	5.13	4	align flex accumulator	N	N	
OP 3C	5.16	4	Isolate SI pump	Y	N	backup actions to CR only
OP 3C	5.23.6	5	Align containment purge	N	N	
OP 3C	5.23.7	4	align flex N2	N	N	Restraint for entry into Mode 5 only, plant is stable and can remain in this condition until area accessible
OP 3C	attachment A	3/4/5	borate to refueling concentration	Y	N	backup to CR only
Stop at step 5.24 because plant is now in mode 5.						

Procedure:	Procedure Step:	Mode:	Action:	Backup to CR or automatic action?	Vital to plant ops	Location of vital action:
OP 7A	5.1.3a	4	adjust CC cooling to RHR	N	Y	Pipeway 2/3, 8' elev.
OP 7A	5.1.4	4	caution tag 851 871s	N	Y	C-59 area
OP 7A	5.1.5	4	align RHR suction	N	Y	C-59 area, -5ft
OP 7A	5.1.6	4	shut RH-716	N	N	
OP 7A	5.2.16	4, 5	CCW temp control	N	Y	CCW HX Room
OP 5D P4	5.2	3	Initiate primary degas	N	Y	primary sample room
OP 5D P4	5.3.3	3	tag shut CV-261c	N	Y	VCT area
OP 5D P4	5.3.5	3	isolate H2 to vct	N	Y	VCT area

Resulting Tables used in EALs RA3 and HA5 are shown below:

Table R-2 SAFE OPS, S/D, C/D AREAS	
Area/Building	**MODE**
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B-32 MCC Area	4

Table H-2 SAFE OPS, S/D, C/D AREAS	
Area/Building	**MODE**
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 /4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B32 MCC Area	4

This validation document (V29) is not used

Results and Conclusions:

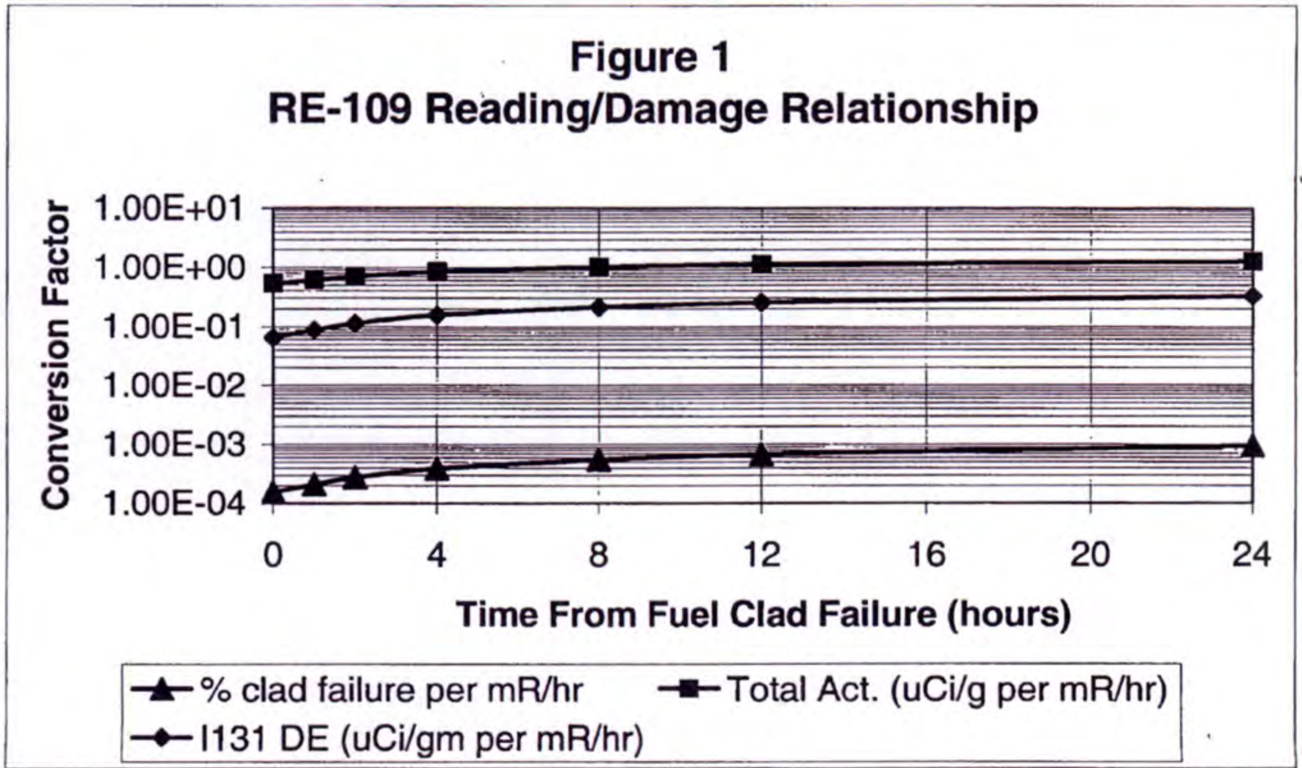
The results were calculated using a source term corresponding to a reactor thermal power operating level of 1683 MWth. The results are applicable generally to other thermal power operating levels because core activity and photon intensity are directly proportional to the reactor thermal operating power level.

The failed fuel monitor reading at time equal to 0 that would correspond to 300 μCi/g I-131 DE is approximately 4500 mR/hr and the failed fuel monitor reading at time equal to 0 that would correspond to 50 μCi/g I-131 DE is approximately 750 mR/hr.

The calculated RCS total radionuclide activity and fuel clad failure percentage correlations for use with the failed fuel monitor (RE-109) reading are graphically displayed in Figure 1 below. To determine fuel clad failure percentage, total primary system activity, or total I-131 DE activity, multiply the respective conversion factor by the failed fuel monitor reading.

Example: An event occurs causing fuel failure. The failed fuel monitor reading 12 hours after fuel failure occurs is 2000 mR/hour. The conversion factor for fuel clad failure from Figure 1 is approximately 7E-04 % clad failure per mR/hr. The estimated fuel clad failure is 7E-04 x 2000 or 1.4 %.

These factors are acceptable for use when the sample system is not isolated, i.e., a containment isolation signal has not occurred. These factors can be used after sample system isolation has occurred, however, the values will only reflect the condition of the primary system at the time of isolation.



3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT Xe-133 specific activity shall be within limits:

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	------------------Note------------------ LCO 3.0.4.c is applicable. --	
	A.1 Verify DOSE EQUIVALENT I-131 ≤50 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT Xe-133 not within limit.	------------------Note------------------ LCO 3.0.4.c is applicable. -- B.1 Restore DOSE EQUIVALENT Xe-133 to within limit.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 >50 μCi/gm.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	---------------------------NOTE-------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT Xe-133 Specific Activity $\leq$ 300 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	---------------------------NOTE-------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 0.5 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of $\geq$ 15% RTP within a 1 hour period

Excerpt from NEI 99-01 Rev6:

EAL SU4 (SU5 PBNP)

Developer Notes:

EAL #1 (unidentified leakage) – For the site-specific leak rate value, enter the higher of 10 gpm or the value specified in the site's Technical Specifications for this type of leakage.

PBNP Tech Specs limit for unidentified leakage is 1 gpm; therefore 10 gpm is entered as specified above

EAL #2 (identified leakage) – For the site-specific leak rate value, enter the higher of 25 gpm or the value specified in the site's Technical Specifications for this type of leakage.

PBNP Tech Specs limit for identified leakage is 10 gpm; therefore 25 gpm is entered as specified above

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

EMERGENCY OPERATING PROCEDURE (EOP) SETPOINTS

PNBP STPT	ERG STPT	PARAMETER	DESCRIPTION	SETPOINT Unit 1	SETPOINT Unit 2	CALC REFERENCE
V.39		Shutdown Margin	Shutdown margin for RCS temperature at 70°F	1% Δk/k	1% Δk/k	2010-0028
V.40		RCP Seal dP	Minimum RCP seal differential pressure for normal RCP seal cooling flow	20 in w.c.	20 in w.c.	2010-0028
V.41		Main Steam Pressure	Minimum pressure differential between intact and ruptured SGs	250 psid	250 psid	2010-0028
V.42		EDG Frequency	Maximum allowable emergency diesel generator frequency	60.3 Hz	60.3 Hz	2010-0028
V.43		D-01 / D-02 Bus Voltage	D-01 / D-02 minimum battery bus voltage	115 Vdc	115 Vdc	2010-0028
V.44			SPARE			2010-0028
V.45			SPARE			2010-0028
V.46	X.04	No. of SGs	Number of steam generators necessary to maintain RCS pressure low enough to gravity feed.	1	1	2010-0028
W.1		RHR Pump Discharge pressure	Minimum RHR pump discharge pressure to maintain pump flow less than 2200 gpm to ensure pump does not reach runout conditions	120 psig	120 psig	2010-0028
W.2			SPARE			2010-0028
Z.1		RCS Temperature (Core Exit)	Core exit temperature indicative of inadequate core cooling (ERG value given)	1200°F	1200°F	2010-0017
Z.2	G.03	RCS Temperature (Core Exit)	Temperature corresponding to 670°F plus normal channel accuracy and post-accident transmitter errors or 700°F, whichever is greater	700°F	700°F	2010-0017
Z.3		PZR Level	Pressurizer level range to ensure adequate inventory to accommodate void growth	20% to 30%	20% to 30%	2010-0016

A.1 STATION BLACKOUT (SBO)

A.1.1 STATION BLACKOUT OVERVIEW

Station Blackout is defined as the complete loss of alternating current electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with a turbine trip and the unavailability of the onsite emergency AC power system). A Station Blackout does not involve the loss of available AC power to buses fed by station batteries through inverters. The event is considered to be terminated upon the restoration of power to the essential switchgear buses from any source, including the alternate AC source which has been qualified as an acceptable coping mechanism. A concurrent single failure or design basis accident need not be assumed during a station blackout event (Reference 2 and Reference 18).

The requirements for Station Blackout are established in 10 CFR 50.63 (Reference 1), which was formally issued in 1988. Guidance for compliance with the regulatory requirements is presented in NUMARC 87-00, Revision 0 (Reference 2) and Regulatory Guide 1.155 (Reference 3). The NRC has not endorsed Revision 1 to NUMARC 87-00, but has accepted specific supplements to NUMARC 87-00 Rev. 0, as described in Appendix K of NUMARC 87-00 Rev. 1 (Reference 2).

The station blackout regulation requires determination of the coping duration category based on criteria provided in Reference 2 and Reference 3. The "required coping duration" is defined as the time between the onset of station blackout and the restoration of off-site AC power to safe shutdown buses. "Coping duration category" is a quantification of the relative risk of a particular facility to the occurrence of a station blackout (loss of all onsite and offsite AC power).

The determination of the required coping duration category is based on several factors, such as the plant design and the probability of severe weather conditions in the area. Once the required coping duration category has been established, the design approach to coping with the station blackout event is demonstrated. This design approach may choose to take credit for either an available alternate AC power source or opt for an AC power-independent design. The plant systems must have the necessary capacity and capability to ensure the core is cooled and containment integrity is maintained for the required station blackout coping duration.

The coping duration categories are 2, 4, 8, or 16 hours, as determined from Table 3-8 of Reference 2. The intent of the regulation is for all domestic nuclear plant sites to fall in either the 2-hour or the 4-hour coping duration category, and then select either the "Alternate AC" or "AC-Independent" coping methodology for their specific plant. The NRC bases for coping duration category objectives are described in Section 2.3.2 of Reference 2. The major contributor to overall station blackout risk is the likelihood of losing off-site power and the duration of power unavailability. The stated objective of the NRC is to reduce the core damage frequency due to station blackout to approximately 10^{-5} per year for the average site. This objective is accomplished by requiring either a four hour coping capability or use of an Alternate AC (AAC) source.

PBNP's original response to the SBO rule concluded that the required coping duration category was 8 hours and used the Gas Turbine Generator (GTG) G-05 as the sole Alternate AC (ACC) source to power the safe shutdown loads of both blacked out units. Because the GTG cannot be

shown to be available within 10 minutes of the onset of station blackout, a one hour coping assessment was performed as required by Section 7.1.2 of Reference 2. (Reference 4 and Reference 8)

The coping duration category was subsequently revised to 4 hours based on a change in the extremely severe weather (ESW) group classification as discussed in Section A.1.2. With the addition of the G-03 and G-04 EDGs, the SBO minimum redundancy requirements of emergency AC (EAC) power supplies for normal safe shutdown of both units is exceeded and utilization of an EDG as an AAC source is allowed. By definition, a unit with an available EAC power supply is not blacked out. However, any EDG credited as an AAC source must be capable of handling the safe shutdown loads in both the blacked out and non-blacked out units (Reference 2). The PBNP EDGs meet this requirement. Therefore, the present coping methodology utilizes the Gas Turbine Generator (GTG) G-05 or an Emergency Diesel Generator (EDG) from the non-blacked out unit as Alternate AC (ACC) sources. An EDG will start, accelerate to rated frequency and voltage, and can be connected to an EAC bus in either unit within ten minutes of SBO initiation. The GTG will be manually started, accelerate to rated frequency and voltage, and be available to power the safe shutdown loads within one hour of SBO initiation (Reference 6, Reference 15). Since PBNP continues to use the GTG as one of the ACC sources, and it cannot be shown to be available within 10 minutes, the one hour coping assessment has been retained and is described in Section A.1.3.

A.1.2 STATION BLACKOUT COPING DURATION CATEGORY DETERMINATION

The potential for long duration loss of off-site power (LOOP) events can have a significant impact on station blackout risk and required coping duration. Long duration LOOP events are typically associated with grid failures due to severe weather conditions or unique transmission system features. Shorter duration LOOP events tend to be associated with plant specific switchyard features. Per Reference 1, the required coping duration shall be based on the following factors:

1. The redundancy of the emergency standby power system
2. The reliability of each of the emergency power sources
3. The expected frequency of a loss of offsite power
4. The probable time required to restore offsite power

Offsite Power Design Characteristic Group

The regulatory guidance (Reference 2, Tables 3-5a and 3-6a; Reference 3, Table 4) has established three basic groups (P1, P2, and P3) for categorizing the design of the preferred offsite power system. A category of P3 is assigned to those plants with a frequency of grid-related loss of offsite power events greater than once in 20 site-years, which is limited to St. Lucie, Turkey Point and Indian Point (Reference 2).

Since PBNP is not included among the three noted plant sites, further evaluation of several factors is necessary to establish the Offsite Power Design Characteristic Group. The applicable group is defined based on combinations of the following three factors:

- extremely severe weather
- severe weather
- offsite power system independence

Extremely Severe Weather (ESW Group)

The estimated frequency of loss of offsite power due to extremely severe weather is determined by the annual expectation of storms at the site with wind velocities equal to or greater than 125 mph. These events are normally associated with the occurrence of hurricanes where high windspeeds may cause widespread transmission system unavailability for extended periods. Since electrical distribution systems are not designed for such conditions, it is assumed the occurrence of such windspeeds will directly result in the loss of offsite power.

The estimated frequency may be determined based on either site-specific data or on data from local weather stations. Table 3-2 of Reference 2 summarizes site-specific National Oceanic Atmospheric Administration (NOAA) data for the estimated frequency of occurrence of extremely severe weather. As published in this table, PBNP has an event frequency of 0.0036, and therefore was categorized in ESW Group 4 (Reference 4). Subsequent review determined the NOAA data for extremely severe weather was overly conservative for PBNP, and that an ESW event frequency supporting an ESW Group 2 category was justified (Reference 5). This departure from the NUMARC 87-00 criteria was reviewed and approved by the NRC (Reference 6).

Severe Weather (SW Group)

Table 6 of Reference 3 and Part 3.2.1.C of Reference 2 define the severe weather factor based on the frequency of a loss of offsite power due to severe weather. The severe weather considered includes snow, tornadoes, high winds, and storms with salt spray. These are related by the equation:

$$\text{frequency} = 1.3 \times 10^{-4} \times h_1 + b \times h_2 + 0.012 \times h_3 + c \times h_4$$

The variables in this equation are defined for PBNP in Reference 2, Section 3.2.1.C:

h_1 = annual expectation of snowfall for site, in inches; this is 42.0 inches for PBNP

h_2 =annual expectation of tornadoes with windspeeds greater than or equal to 113 miles per hour, in events per square mile; this is 0.000035 for PBNP

h_3 =annual expectation of storms with wind velocities between 75 and 124 mph; this is 0.1 for PBNP

h_4 =annual expectation of storms with significant salt spray for the site; this is 0.0 for PBNP.

b =72.3; the PBNP offsite power system design connects four 345 kV transmission circuits to the plant switchyard via a single right-of-way.

c =0; the PBNP site is not considered vulnerable to the effects of salt spray.

These factors, when combined in the severe weather frequency equation, yield an estimated frequency of loss of offsite power due to severe weather of 0.0092. This places PBNP in SW Group 2.

Independence of the Offsite Power System (I Group)

Reference 3, Table 5, defines the offsite power system independence factor, and Reference 2 Section 3.2.1.D simplifies the determination:

If: (a) all offsite power sources are connected to the safe shutdown buses through one switchyard or through multiple electrically connected switchyards, and (b1) the normal power source is from the main generator and there are no automatic and one or more manual transfers of all safe shutdown buses to the preferred or alternate offsite power sources, or (b2) there is one automatic and no manual transfers of the safe shutdown buses to one preferred or one alternate offsite power source, the site falls in the I-3 group. Otherwise, the site is assigned to the I-1/2 group.

The I-1/2 group is characterized by features associated with greater independence and redundancy of sources, and a more desirable transfer scheme. I-3 sites have simpler, less desirable offsite power systems and switchyard capabilities.

Condition a: The PBNP offsite power system consists of four (4) 345 kV transmission circuits, connected via a single right-of-way, to a single switchyard which serves both PBNP units. On this basis, the answer to Condition A is considered to be "YES" for the PBNP site.

Condition b1 and b2: The PBNP auxiliary power distribution system provides offsite power connections to the safety-buses of each unit via the high voltage station auxiliary transformers and the low voltage station auxiliary transformers. This normal supply of power to the safety-related buses is derived from offsite power sources. Upon loss of the preferred offsite power source to the safety-related buses of one unit, the buses will be powered from the preferred power source of the other unit. On this basis, the answers to both Condition b(1) and b(2) are considered to be "NO", and the PBNP site is classified in the I-1/2 Group.

Offsite AC Power Design Characteristic Group Determination

The combination of the ESW, SW and I factors results in an Offsite Power Design Characteristic Group of P1 for PBNP, based on Reference 3, Table 4.

Emergency AC Power Configuration Group

Regulatory guidance defines four Emergency AC (EAC) Power Configuration groups (A, B, C, and D) based on the availability and redundancy of the emergency power supplies. Reference 2 Section 3.2.2 clarifies the EAC groups, basing it on the number of EAC power supplies required to handle the safe shutdown loads and on the number of additional EAC power supplies available. The PBNP EAC power configuration group is C, based on the following:

PBNP is a two-unit site with four shared Emergency Diesel Generators (EDGs) and one gas turbine generator (GTG). The two Train A EDGs are identical components with a 2000 hour rated output of 2850 kW at 4.16 kV. The two Train B EDGs are identical components with a 2000 hour rated output of 2848 kW at 4.16 kV. All four EDGs are available to support the safe shutdown equipment of either PBNP unit, and a single EDG can supply adequate power to the safe shutdown loads in both units. The GTG has a rating of 23.10 MVA at an output voltage of 13.8 kV, and can supply adequate power to the safe shutdown loads in both units.

Therefore, because only one EDG is necessary to operate safe shutdown equipment for both units following a loss of offsite power, the EAC power configuration group at PBNP is "C", as a 1 out of 2 EDG, dedicated, or 1 out of 3 EDGs, shared configuration per Table 3-7 of Reference 2. Additionally, the PBNP SBO licensing basis permits the use of either the GTG or an EDG as the AAC source.

Target Standby Diesel Generator Reliability

The reliability of the EAC power sources has a key role in the quantification of risk due to SBO. A target value for reliability was therefore made a factor in establishing the required SBO coping duration. The EDG target reliability was selected to be 0.975 based on the original EAC configuration group determination of "D" (i.e., prior to the installation of G-03 and G-04) and the reliability data that existed at the time of the initial SBO evaluation. (Reference 4) These reliability computations utilized the NRC-recommended methodology of EPRI Report NSAC-108 (Reference 7)

Because PBNP offsite power design group is P1, and EAC configuration is C, the target EDG reliability value may be 0.950 or 0.975 per Table 2 of Reference 3. PBNP has retained the reliability target value of 0.975 (Reference 15). PBNP has implemented an EDG reliability program which is based on the methodology of EPRI Report NSAC-108 and conforms to the guidance of RG 1.155, Position 1.2 (Reference 8 and Reference 15).

Coping Duration Category Determination Summary

The previous determinations are summarized below:

Offsite AC Power Design Characteristic Group	=P1
Emergency AC Power Configuration Group	=C
Target Standby Diesel Generator Reliability	=0.975

In accordance with Table 3-8 of Reference 2 and Table 2 of Reference 3, the group determinations listed above result in a coping duration category for PBNP of four hours.

A.1.3 STATION BLACKOUT COPING ANALYSES

Condensate Inventory for Decay Heat Removal

This analysis ensures that PBNP has sufficient condensate inventory to support the decay heat removal function for the SBO event duration. Section 7.2.1 of Reference 2 provides a simplified calculation approach to determine the required condensate volume. This analysis is satisfied by demonstrating that Technical Specification volume requirements envelop the volume estimated by the Reference 2 methodology.

At a core power of 1800 MWt, 14,000 gallons of condensate water are required for the one hour SBO event duration based on the methodology of Reference 2. However in order to maintain the same margin set by the NRC in Reference 8 for subsequent switchover to the long-term AFW water supply, the minimum CST usable volume is set at 15,410 gallons. This volume is bounded by the Technical Specification CST volume requirements which includes additional margin to

1-PT-FP-001

UNIT 1 CONTAINMENT FIRE DETECTOR TEST (REFUELING)

DOCUMENT TYPE:	Technical
CLASSIFICATION:	Safety Related
REVISION:	3
REVIEWER:	N/A
APPROVAL AUTHORITY:	Department Manager
PROCEDURE OWNER (title):	Group Head
OWNER GROUP:	Operations

Verified Current Copy: ______________________ ____________ ____________

Signature Date Time

List pages used for Partial Performance

____

____

Controlling Work Document Numbers

____________ ____________

____________ ____________

Completed Procedure Review:

______________________ ______________________ ____________ ____________

Shift Supervision (Print) Shift Supervision (Sign) Date Time

TABLE OF CONTENTS

CFP_________

1.0 PURPOSE

1.1 This test is to verify the integrity and standby operability of the Unit 1 containment smoke detection system.

1.2 This test is required by NFPA 72, National Fire Alarm Code and NEIL Member's Manual.

2.0 PREREQUISITES

NOTE: The steps in the prerequisites section may be performed in any sequence.

2.1 The fire detection system including the fireworks stations are available for testing the detectors.

2.2 Radiation Protection has been notified of the areas of containment that will be accessed and Radiation Work Permit has been obtained.

2.3 Special Tools and Equipment

- Smoke Detector Test Pole - Solo 301-024 or equivalent (I&C M&TE Locker)
- Smoke Detector Tester - Stock 915-2593, 2 cans (OPS Flammable Locker)
- Extension Ladder
- Two radios

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Any testing performed greater than or equal to 6 feet above floor level requires the use of personnel safety equipment (e.g., safety belt, etc.).

3.2 Detectors that have alarmed will **NOT** alarm again after acknowledging and are considered Out-Of-Service till Reset. Alarms should be reset prior to exiting containment.

3.3 Any deficiencies should be immediately reported to the Shift Management for appropriate action in accordance with OM 3.27.

INITIALS

4.0 INITIAL CONDITIONS

NOTE: ALARA shall be considered at all times

4.1 Unit 1 is in any condition. ______

4.2 Radiation Protection has been notified to provide support, if required. ______

4.3 A pre-job brief has been performed. ______

4.4 This test is being done to satisfy: ______

_____ The normally scheduled callup. Task Sheet No. ______________ ______

NOTE: If this test is being performed to satisfy PMT or off-normal frequency requirements, Shift Management may N/A those portions of the procedure that are NOT applicable for the performance of the PMT. The use of N/A is NOT acceptable for Initial Conditions, Precautions and Limitations, or procedure steps that pertain to the equipment requiring PMT, nor is it acceptable for restoration of equipment/components unless the component has been declared inoperable.

_____ Post maintenance operability test
Equipment ID _____________
WO No(s). ________________
Task Sheet No.(s) ________________
_____ Special test - no numbers
Explain: ______________________________________

4.5 **Permission to Perform Test**

The conditions required by this test are consistent with required plant conditions, including equipment operability. Permission is granted to perform this test.

__________________________________ ____________ ____________
Shift Management Date Time

NOTE: The first detector smoked will require a longer smoke time since the detector will first go into the Alert Activation condition before the Alarm Activation.

NOTE: The smoker of the detector should <u>NOT</u> announce the detector to be tested. After the detector is smoked, the individual at the Fireworks station will call out the detector in alarm and the smoker will confirm it.

5.0 <u>PROCEDURE</u>

NOTE: Annunciator Window 1C20 A 1-2, FACP ALARM, will be received with the first detector tested.

5.1 Ensure Control Room has been informed of the fire alarms that will come in during test performance.

5.2 At the Fireworks station, go to the sitemap screen to begin the test for a given elevation.

NOTE: Attachment B may be referenced for detector location.

5.3 Using the Smoke Pole, smoke a detector.

5.4 **<u>WHEN</u>** the detector goes into Alarm Activation status with the outlined area blinking in red,
<u>THEN</u> perform the following:

5.4.1 Control Room Operator Acknowledge Annunciator Window 1C20 A 1-2, FACP ALARM.

NOTE: For each elevation, it is only necessary to drill down to the appropriate Containment elevation screen for the first detector tested.

5.4.2 On the Containment screen that shows the individual detectors, confirm detector in alarm.

5.4.3 Touch or click on "ACK" in the bottom right hand corner.

5.4.4 To confirm the detector, radio the detector smoker the detector that went into alarm.

5.4.5 On Attachment A, initial for the detector that went into alarm.

NOTE: The preferred means of testing is to reset the alarms after all detectors have been tested. Resetting detectors will require the first detector after resetting to go through the Alert Activation condition slowing down the testing.

5.5 Repeat Steps 5.3 through 5.4.5 for a given elevation.

5.6 **WHEN** an elevation is complete,
THEN repeat Steps 5.2 through 5.4.5.

5.7 **WHEN** all the detectors have been tested,
THEN ensure the detectors are reset on the control room Fireworks PC using the reset in the bottom left hand corner of the touch screen.

6.0 ACCEPTANCE CRITERIA

6.1 All the detectors tested in Attachment A went into the alarm activation condition when smoked.

7.0 REFERENCES

7.1 NFPA 72, National Fire Alarm Code

7.2 NEIL Member's Manual

7.3 OI 40A, Fire Alarm Control Panel and Fireworks PC Operation

7.4 FPPDD, NFPA 805 Fire Protection Program Design Document

7.5 OM 3.27, Control of Fire Protection & NFPA 805 Equipment

7.6 Drawings

7.6.1 EST FPE-001

7.6.2 EST FPE-002

7.6.3 EST FPE-003

8.0 BASES

None

REMARKS: ____

ATTACHMENT A
DETECTOR RESPONSE

Detector	Location	Acceptance Criteria	Initials	Notes
XS-5175	8'	Alarmed		
XS-5176	8'	Alarmed		
XS-5177	8'	Alarmed		
XS-5178	8'	Alarmed		
XS-5179	8'	Alarmed		
XS-5180	8'	Alarmed		
XS-5181	8'	Alarmed		
XS-5182	8'	Alarmed		
XS-5183	8'	Alarmed		
XS-5184	8'	Alarmed		
XS-5185	21'	Alarmed		
XS-5186	21'	Alarmed		
XS-5187	21'	Alarmed		
XS-5188	21'	Alarmed		
XS-5189	21'	Alarmed		
XS-5190	21'	Alarmed		Above Blowdown Isolation
XS-5191	21'	Alarmed		Above Blowdown Isolation
XS-5192	21'	Alarmed		Above Fan Cooler 1W1C1 & 2
XS-5193	21'	Alarmed		Above Fan Cooler 1W1C1 & 2
XS-5194	21'	Alarmed		
XS-5195	21'	Alarmed		
XS-5196	21'	Alarmed		
XS-5197	21'	Alarmed		Above Fan Cooler 1W1D1 & 2
XS-5198	21'	Alarmed		
XS-5199	21'	Alarmed		
XS-5200	21'	Alarmed		
XS-5201	21'	Alarmed		
XS-5202	21'	Alarmed		
XS-5203	21'	Alarmed		
XS-5204	21'	Alarmed		From 46' access to Incore Detector
XS-5205	46'	Alarmed		
XS-5206	46'	Alarmed		
XS-5207	46'	Alarmed		Top access from RCP Cubicle
XS-5208	46'	Alarmed		
XS-5209	46'	Alarmed		
XS-5210	46'	Alarmed		
XS-5211	46'	Alarmed		
XS-5212	46'	Alarmed		RCP cubicle top access
XS-5213	46'	Alarmed		
XS-5214	46'	Alarmed		

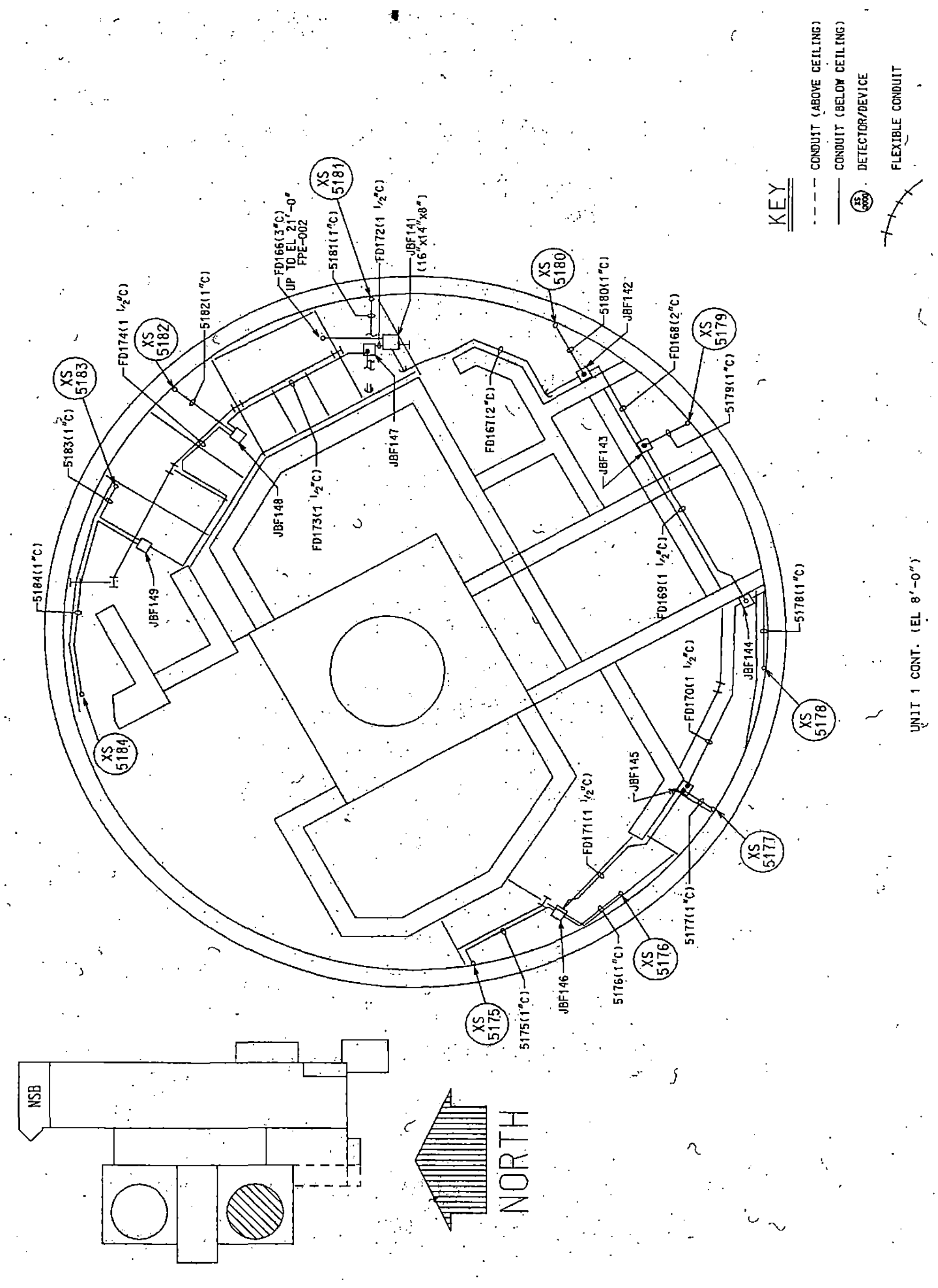
Acceptance Criteria Satisfied for all tested detectors in Attachment A.

(Circle one) SAT UNSAT

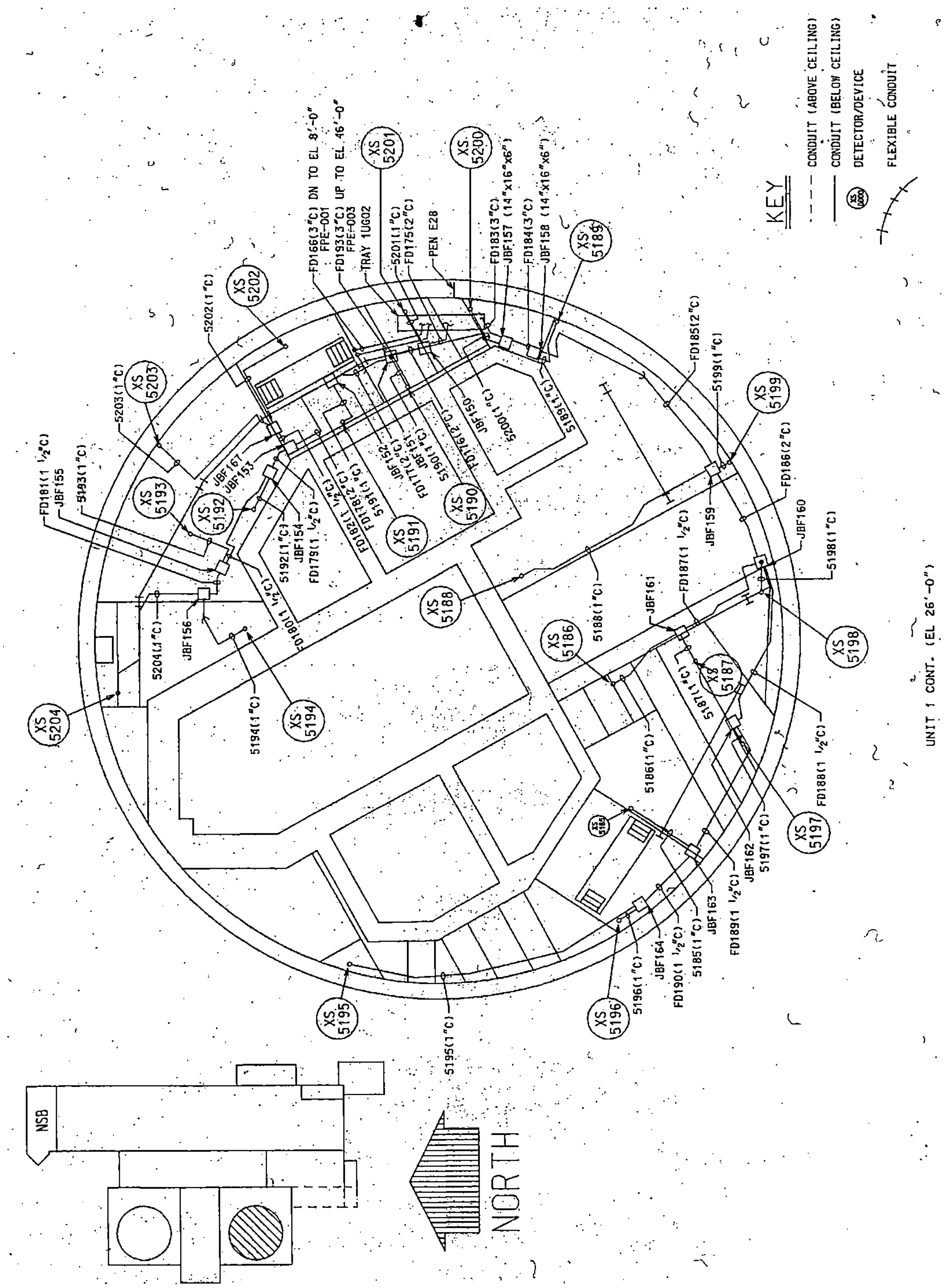
_________________ / _________________ _________________ / _________________

Performer Date/Time SRO Date/Time

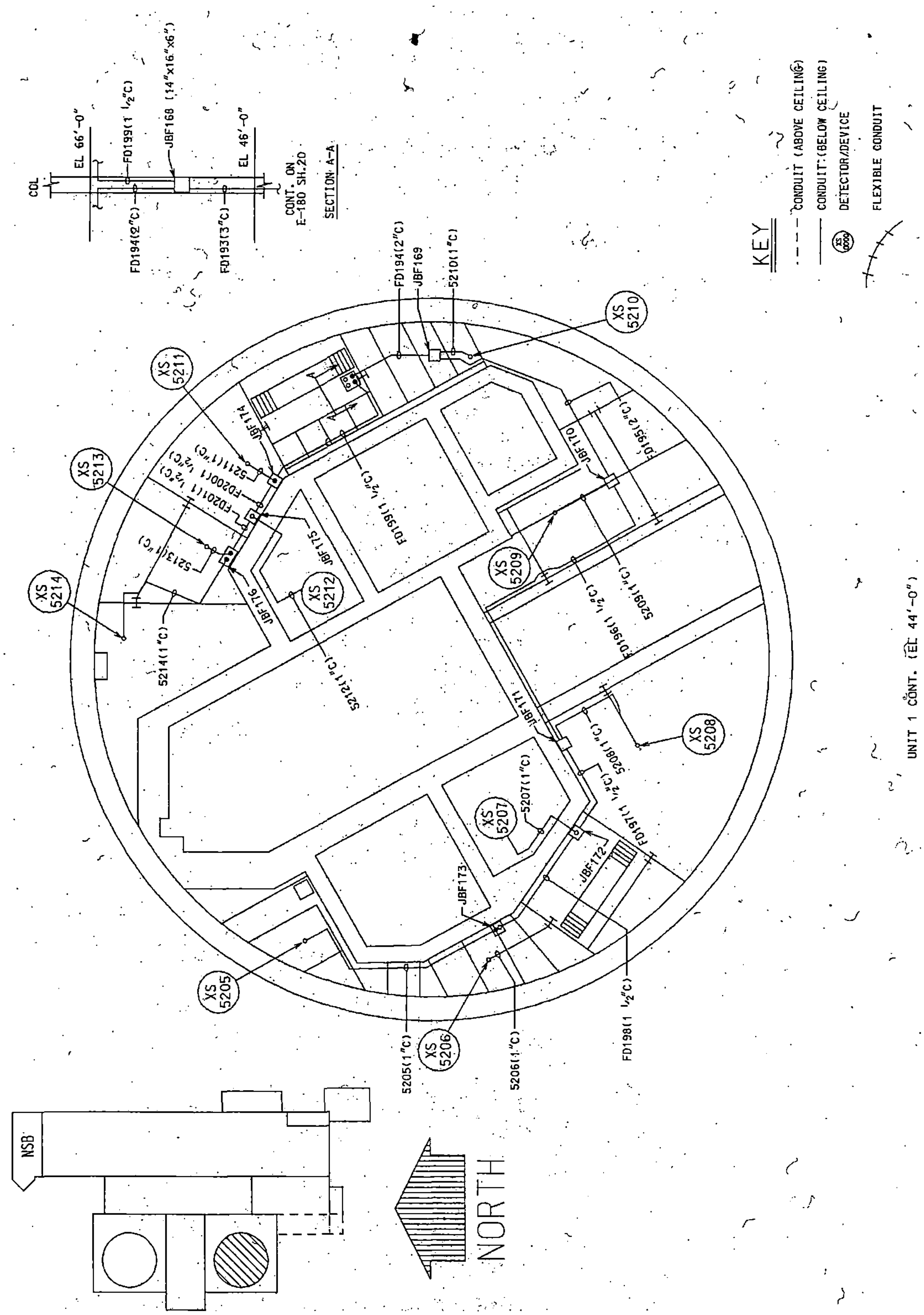
ATTACHMENT B
DETECTOR LAYOUT



UNIT 1 CONTAINMENT FIRE DETECTOR TEST
(REFUELING)



UNIT 1 CONTAINMENT FIRE DETECTOR TEST (REFUELING)



A. PURPOSE

1. This procedure provides directions to respond to a loss of all AC safeguards power.
2. This procedure is applicable for initiating events occurring in MODES 1, 2, 3, and 4.

B. SYMPTOMS OR ENTRY CONDITIONS

1. The symptom of a loss of all AC power is the indication that both AC safeguards trains are deenergized.
2. This procedure is entered from:
   - EOP-0 UNIT 1, REACTOR TRIP OR SAFETY INJECTION, Step 3, on indication that both AC safeguards trains are deenergized.
   - This procedure may be entered directly, on indication that both AC safeguards trains are deenergized.

C. REFERENCES

1. EC 283586, Transition to 10 CFR 50.48(c) - NFPA 805 From App R
2. EPM Report R2168-1003C-001

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Verify Reactor Trip:** • Reactor trip and bypass breakers - OPEN • Neutron flux-LOWERING	Manually trip reactor: • Train A • Train B
2	**Verify Turbine Trip:** a. Turbine stop valves-BOTH SHUT	a. Manually trip turbine. **IF** Turbine will not trip, **THEN** SHUT MSIVs.

NOTES

- **Foldout page shall be monitored throughout this procedure.**
- **CSF status trees should be monitored for information only. CSPs should not be implemented.**

3 **Secure RCPs**

a. Ensure both RCPs - STOPPED

- 1P-1A
- 1P-1B

b. Place steam dump mode control - MANUAL

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Check If RCS Is Isolated:**	
	a. PZR PORVs - BOTH SHUT • 1RC-430 • 1RC-431C	a. **IF** PZR pressure is less than 2335 psig, **THEN** manually shut PORVs.
	b. Letdown orifice outlet valves - SHUT • 1CV-200A • 1CV-200B • 1CV-200C	b. Manually shut valves.
	c. Letdown containment isolation valves - SHUT • 1CV-371A • 1CV-371	c. Manually shut valves.
	d. RCP seal return isolation valve - SHUT • 1CV-313A	d. Manually shut valve.
	e. RCS sample valves - SHUT • 1SC-966A, PZR steam space sample containment isolation valve • 1SC-966B, PZR liquid space sample containment isolation valve • 1SC-966C, RCS hot leg sample containment isolation valve	e. Manually shut valves.
	f. Head vent system - ENERGIZED	f. Go to Step 5.
	g. Reactor vessel head vent solenoids - SHUT • 1RC-570A • 1RC-570B	g. Manually shut valves.
	h. PZR vent valves - SHUT • 1RC-580A • 1RC-580B	h. Manually shut valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Verify AFW Flow - GREATER THAN OR EQUAL TO 230 gpm.**	Perform the Following: a. Ensure TDAFW Pump steam supply MOV's OPEN: • 1MS-2020 • 1MS-2019 b. Ensure TDAFW Pump discharge MOV's OPEN: • 1AF-4000 • 1AF-4001 c. **IF** AFW **NOT** established **THEN** perform ATTACHMENT K, ESTABLISHING HEAT SINK while continuing with this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	**TRY TO RESTORE POWER TO ANY SAFEGUARDS BUS**	
	a. Emergency Diesel Generators - All Running: • G-01, train A • G-02, train A (alternate) • G-03, train B • G-04, train B (alternate)	a. Try to start non-running EDGs 1) Ensure diesel mode selector switch in AUTO. 2) Place control switch to START. 3) Ensure generator field flash occurs. 4) Ensure green READY TO LOAD light is energized. 5) **IF** NO diesel is running, **THEN** go to Step 10.
	b. AC safeguards Buses - AT LEAST ONE TRAIN ENERGIZED o 1A-05 **AND** 1B-03 OR o 1A-06 **AND** 1B-04	b. **IF** **NO** 4160v SAFEGUARDS BUS is energized, **THEN** go to STEP 7. 1) **IF** Bus 1A-05 is energized, **THEN** energize 1B-03 by closing: • 1A52-58 • 1B-16B Go to Step 9 2) **IF** Bus 1A-06 is energized, **THEN** energize 1B-04 by closing: • 1A52-84 • 1B52-17B Go to Step 9
	c. Go to Step 9	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

In fire scenarios, G-02 is susceptible to automatic tripping on overload when it is supplying both 1A-05 and 2A-05 due to spurious equipment operations.

7 **Restore Power to "A" Safeguards Bus:**

a. Check G-01 - RUNNING:

1) **IF** G-01 is running **AND** 1A52-60, G-01 to Bus 1A-05 Breaker, is **NOT** closed, **THEN** perform the following:

a) Ensure 1A52-57, 1A-03 to 1A-05 Bus Tie Breaker, is OPEN.

b) Try to auto close breaker by placing control switch to trip position then release.

c) **IF** breaker will NOT auto close, **THEN** perform the following:

1. Place mode selector switch in EXERCISE
2. Turn synch switch ON
3. At C-02, manually CLOSE breaker control switch
4. Turn synch switch OFF

d) **IF** 1A-05 is energized from its normal EDG G-01, **THEN** energize 1B-03 by closing:

- 1A52-58
- 1B52-16B

e) Go to Step 9.

RESPONSE NOT OBTAINED:

a. Check G-02 - RUNNING

1) **IF** G-02 is running **AND** 1A-05 is still **NOT** energized, **THEN** perform the following:

a) Ensure 1A52-57, 1A-03 to 1A-05 Bus Tie Breaker, is OPEN

b) Ensure 1A52-60, G-01 to Bus 1A-05 breaker, is OPEN and in PULLOUT

c) Unlock and place 1A52-66, G-02 to Bus 1A-05 breaker, control switch in auto

d) **IF** 1A52-66, G-02 to Bus 1A-05 breaker, is **NOT** closed, **THEN** perform the following:

1) Try to auto close breaker by placing control switch to trip position then release

2) **IF** breaker will NOT auto close, **THEN** perform the following:

a. Place mode selector switch in EXERCISE
b. Turn synch switch ON
c. At C-02, manually CLOSE breaker control switch
d. Turn synch switch OFF

e) **IF** 1A-05 is energized from its alternate EDG G-02, **THEN** energize 1B-03 by CLOSING:

- 1A52-58
- 1B52-16B

f) Go to Step 9.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

In fire scenarios, G-04 is susceptible to automatic tripping on overload when it is supplying both 1A-06 and 2A-06 due to spurious equipment operations.

8 **Restore Power to "B" Safeguards Bus:**

a. Check G-03 - RUNNING:

1) **IF** G-03 is running **AND** 1A52-80, G-03 to Bus 1A-06 Breaker, is **NOT** closed, **THEN** perform the following:

a) Ensure 1A52-77, 1A-04 to 1A-06 Bus Tie Breaker, is OPEN.

b) Try to auto close breaker by placing control switch to trip position then release.

c) **IF** breaker will NOT auto close, **THEN** perform the following:

1. Place mode selector switch in EXERCISE
2. Turn synch switch ON
3. At C-02, manually CLOSE breaker control switch
4. Turn synch switch OFF

d) **IF** 1A-06 is energized from its normal EDG G-03, **THEN** energize 1B-04 by closing:

- 1A52-84
- 1B52-17B

e) Go to Step 9.

RESPONSE NOT OBTAINED:

a. Check G-04 - RUNNING

1) **IF** G-04 is running **AND** 1A-06 is still **NOT** energized, **THEN** perform the following:

a) Ensure 1A52-77, 1A-04 to 1A-06 Bus Tie Breaker, is OPEN

b) Ensure 1A52-80, G-03 to Bus 1A-06 breaker, is OPEN and in PULLOUT

c) Unlock and place G-04 to Bus 1A-06 breaker 1A52-86 control switch in auto

d) **IF** 1A52-86, G-04 to Bus 1A-06 breaker, is **NOT** closed, **THEN** perform the following:

1) Try to auto close breaker by placing control switch to trip position then release

2) **IF** breaker will NOT auto close, **THEN** perform the following:

a. Place mode selector switch in EXERCISE

b. Turn synch switch ON

c. At C-02, manually CLOSE breaker control switch

d. Turn synch switch OFF

e) **IF** 1A-06 is energized from its alternate EDG G-04, **THEN** energize 1B-04 by CLOSING:

- 1A52-84
- 1B52-17B

f) Go to Step 9.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	**Verify One Train Of Safeguards Bus Energized:**	
	a. AC safeguards Buses - AT LEAST ONE TRAIN ENERGIZED o 1A-05 **AND** 1B-03 OR o 1A-06 **AND** 1B-04	a. Try to energize AC safeguards Bus from Control Room using any available power source. Go to Step 10.
	b. Monitor running EDG status: • Check frequency on running diesels - BETWEEN 59.7 Hz AND 60.3 Hz • Check voltage on running diesels - BETWEEN 4050 Vac AND 4300 Vac	b. Perform the following: • Adjust governor to establish 60 Hz • Adjust voltage regulator to establish 4160 VAC **IF** EDG cooling, voltage, or frequency can **NOT** be maintained, **THEN** shutdown affected EDG(s) 1) Trip affected EDG(s) and place output breaker in pull out: o G-01, 1A52-60 o G-02, 1A52-66 o G-03, 1A52-80 o G-04, 1A52-86 2) Place mode selector switch for affected EDG(s) to LOCAL 3) **IF** any EDG can **NOT** be shut down from the Control Room, **THEN** locally push affected EDG(s) engine stop push button
	c. Service water header pressure GREATER THAN OR EQUAL TO 50 psig	c. Manually start pumps and align valves as necessary to establish service water header pressure greater than or equal to 50 psig.

(Step 9, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 9. continued from previous page)

d. Trip and close contactor(s) for tripped battery chargers aligned to supply DC Buses

- o D-07
- o D-08
- o D-09
- o D-107
- o D-108
- o D-109

e. Return to procedure and step in effect and implement CSPs as necessary

CAUTION

Local manual closure of breakers bypasses all lockouts and interlocks and may result in equipment damage.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	**Check 1A-03:** • No Lockouts	Go to Step 28.
11	**Check 1A-05:** • No Lockouts	Go to Step 28.
12	**Check 1X-04** • No Lockouts • Power Available o G-05 GTG o Offsite	Go to Step 18.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13	**Restore Power To 13.8 kV:**	
	a. Check bus H-02 - ENERGIZED	a. Perform the following: **IF** Bus H-01 is energized **OR** Bus H-03 energized, **THEN**: 1) Ensure Bus H-02 Normal Feed - OPEN • H52-20 2) Ensure H-03 to H-01 Bus Tie Breaker -CLOSED • H52-31 3) CLOSE H-02 to H-01 Bus Tie Breaker. • H52-21 **IF** Bus H-01 is **NOT** energized, **THEN** start G-05 Gas Turbine per ATTACHMENT E, POWER RESTORATION USING GAS TURBINE, Continue with Step 18.
14	**Energize Bus 1A-03 From 1X-04**	Go to Step 18.
	a. Reset and CLOSE Bus H-02 feed to 1X-04 • H52-22	
	b. Ensure 1A-03 to 1A-01 Bus Tie Breaker - OPEN • 1A52-37	
	c. Reset and CLOSE Bus 1A-03 Normal Feed • 1A52-36	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15	**Energize Bus 1A-05 From 1A-03**	Go to Step 28.
	a. Ensure G-01 to Bus 1A-05 breaker OPEN • 1A52-60	
	b. Ensure G-02 to Bus 1A-05 breaker OPEN • 1A52-66	
	c. Turn on synchronizing switch for 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57	
	d. Trip and close 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57	
	e. Turn off synchronizing switch for 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57	
16	**Check 480 V Safeguard Bus 1B-03 - ENERGIZED**	Perform the following: a. Close Bus 1A-05 feed to 1X-13 • 1A52-58 b. Close Bus 1B-03 Normal Feed • 1B52-16B **IF** 1B-03 **NOT** energized, **THEN** go to Step 28.
17	**Go to Step 46.**	
18	**Check 1X-02** • No Lockouts • Power Available	Go to Step 23.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19	**Energize Bus 1A-03 From 1A-01** a. Ensure Bus 1A-03 Normal Feed - OPEN • 1A52-36 b. Ensure 1A-03 to 2A-03 Bus Tie Breaker - OPEN • 1A52-40 c. Turn synch switch for 1A-03 to 1A-01 Bus Tie Breaker - ON • 1A52-37 d. Close 1A-03 to 1A-01 Bus Tie Breaker • 1A52-37 e. Turn synch switch for 1A-03 to 1A-01 Bus Tie Breaker - OFF • 1A52-37	Go to Step 23.
20	**Energize Bus 1A-05 From 1A-03** a. Ensure G-01 to Bus 1A-05 breaker OPEN • 1A52-60 b. Ensure G-02 to Bus 1A-05 breaker OPEN • 1A52-66 c. Turn on synchronizing switch for 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57 d. Trip and close 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57 e. Turn off synchronizing switch for 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57	Go to Step 26.
21	**Check 480 V Safeguard Bus 1B-03 - ENERGIZED**	Perform the following: a. Close Bus 1A-05 feed to 1X-13 • 1A52-58 b. Close Bus 1B-03 Normal Feed • 1B52-16B **IF** 1B-03 **NOT** energized, **THEN** go to Step 28.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22	Go To Step 46.	
23	**Check 2X-04** • No Lockouts • Power Available	Go to Step 28.
24	**Energize Bus 1A-03 From 2A-03** a. Ensure Bus 1A-03 Normal Feed - OPEN • 1A52-36 b. Ensure 1A-03 to 1A-01 Bus Tie Breaker - OPEN • 1A52-37 c. Turn synch switch for 1A-03 to 2A-03 Bus Tie Breaker - ON • 1A52-40 d. Close 1A-03 to 2A-03 Bus Tie Breaker • 1A52-40 e. Turn synch switch for 1A-03 to 2A-03 Bus Tie Breaker - OFF • 1A52-40	Go to Step 28.
25	**Energize Bus 1A-05 From 1A-03** a. Ensure G-01 to Bus 1A-05 breaker OPEN • 1A52-60 b. Ensure G-02 to Bus 1A-05 breaker OPEN • 1A52-66 c. Turn on synchronizing switch for 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57 d. Trip and close 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57 e. Turn off synchronizing switch for 1A-03 to 1A-05 Bus Tie Breaker • 1A52-57	Go to Step 28.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
26	**Check 480 V Safeguard Bus 1B-03 - ENERGIZED**	Perform the following: a. Close Bus 1A-05 feed to 1X-13 • 1A52-58 b. Close Bus 1B-03 Normal Feed • 1B52-16B **IF** 1B-03 **NOT** energized, **THEN** go to Step 28.
27	**Go To Step 46.**	
28	**Check 1A-04:** • No Lockouts	Go to Step 46.
29	**Check 1A-06:** • No Lockouts	Go to Step 46.
30	**Check 1X-04** • No Lockouts • Power Available o G-05 GTG o Offsite	Go to Step 36
31	**Restore Power To 13.8 kV:**	
	a. Check bus H-02 - ENERGIZED	a. Perform the following: **IF** Bus H-01 is energized **OR** Bus H-03 energized, **THEN**: 1) Ensure Bus H-02 Normal Feed - OPEN • H52-20 2) Ensure H-03 to H-01 Bus Tie Breaker -CLOSED • H52-31 3) CLOSE H-02 to H-01 Bus Tie Breaker. • H52-21 **IF** Bus H-01 is **NOT** energized, **THEN** start G-05 Gas Turbine per ATTACHMENT E, POWER RESTORATION USING GAS TURBINE, Continue with Step 36.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
32	**Energize Bus 1A-04 From 1X-04** a. Reset and CLOSE Bus H-03 feed to 1X-04 • H52-22 b. Ensure 1A-04 to 1A-02 Bus Tie Breaker - OPEN • 1A52-55 c. Reset and CLOSE Bus 1A-04 Normal Feed • 1A52-56	Go to Step 36.
33	**Energize bus 1A-06 From 1A-04** a. Ensure G-03 to Bus 1A-06 breaker OPEN • 1A52-80 b. Ensure G-04 to Bus 1A-06 breaker OPEN • 1A52-86 c. Trip and CLOSE Bus 1A-04 Normal Feed to 1A-06 • 1A52-54 d. Turn on synchronizing switch for 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77 e. Trip and close 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77 f. Turn off synchronizing switch for 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77	Go to Step 46.
34	**Check 480 V Safeguard Bus 1B-04 - ENERGIZED**	Perform the following: a. Close Bus 1A-06 feed to 1X-14 • 1A52-84 b. Close Bus 1B-04 Normal Feed • 1B52-17B **IF** 1B-04 **NOT** energized, **THEN** go to Step 46.
35	**Go to Step 46.**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
36	**Check 1X-02** • No Lockouts • Power Available	Go to Step 41.
37	**Energize Bus 1A-04 From 1A-02** a. Ensure Bus 1A-04 Normal Feed - OPEN • 1A52-56 b. Ensure 1A-04 to 2A-04 Bus Tie Breaker - OPEN • 1A52-52 c. Turn synch switch for 1A-04 to 1A-02 Bus Tie Breaker - ON • 1A52-55 d. Close 1A-04 to 1A-02 Bus Tie Breaker • 1A52-55 e. Turn synch switch for 1A-04 to 1A-02 Bus Tie Breaker - OFF • 1A52-55	Go to Step 41.
38	**Energize bus 1A-06 From 1A-04** a. Ensure G-03 to Bus 1A-06 breaker OPEN • 1A52-80 b. Ensure G-04 to Bus 1A-06 breaker OPEN • 1A52-86 c. Trip and CLOSE Bus 1A-04 Normal Feed to 1A-06 • 1A52-54 d. Turn on synchronizing switch for 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77 e. Trip and close 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77 f. Turn off synchronizing switch for 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77	Go to Step 46.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
39	**Check 480 V Safeguard Bus 1B-04 – ENERGIZED**	Perform the following: a. Close Bus 1A-06 feed to 1X-14 • 1A52-84 b. Close Bus 1B-04 Normal Feed • 1B52-17B **IF** 1B-04 **NOT** energized, **THEN** go to Step 46.
40	**Go to Step 46.**	
41	**Check 2X-04** • No Lockouts • Power Available	Go to Step 46.
42	**Energize Bus 1A-04 From 2A-04** a. Ensure Bus 1A-04 Normal Feed – OPEN • 1A52-56 b. Ensure 1A-04 to 1A-02 Bus Tie Breaker – OPEN • 1A52-55 c. Turn synch switch for 1A-04 to 2A-04 Bus Tie Breaker – ON • 1A52-52 d. Close 1A-04 to 2A-04 Bus Tie Breaker • 1A52-52 e. Turn synch switch for 1A-04 to 2A-04 Bus Tie Breaker – OFF • 1A52-52	Go to Step 46.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
43	**Energize bus 1A-06 From 1A-04** a. Ensure G-03 to Bus 1A-06 breaker OPEN • 1A52-80 b. Ensure G-04 to Bus 1A-06 breaker OPEN • 1A52-86 c. Trip and CLOSE Bus 1A-04 Normal Feed to 1A-06 • 1A52-54 d. Turn on synchronizing switch for 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77 e. Trip and close 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77 f. Turn off synchronizing switch for 1A-04 to 1A-06 Bus Tie Breaker • 1A52-77	Go to Step 46.
44	**Check 480 V Safeguard Bus 1B-04 - ENERGIZED**	Perform the following: a. Close Bus 1A-06 feed to 1X-14 • 1A52-84 b. Close Bus 1B-04 Normal Feed • 1B52-17B **IF** 1B-04 **NOT** energized, **THEN** go to Step 46.
45	**Go to Step 46.**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
46	**Verify One Train Of Safeguards Bus Energized:**	
	a. AC safeguards Buses - AT LEAST ONE TRAIN ENERGIZED o 1A-05 **AND** 1B-03 OR o 1A-06 **AND** 1B-04	a. Perform the following: 1) Locally monitor AFW Pump room temperatures. **IF** AFW pump room temperature rises to 120°F or greater, **THEN** refer to AOP-30, Temporary Ventilation for Vital Areas 2) OBSERVE CAUTIONS PRIOR TO STEP 47 and go to Step 47.
	b. Service water header pressure GREATER THAN OR EQUAL TO 50 psig	b. Manually start pumps and align valves as necessary to establish service water header pressure greater than or equal to 50 psig.
	c. Trip and close contactor(s) for tripped battery chargers aligned to supply DC Buses o D-07 o D-08 o D-09 o D-107 o D-108 o D-109	
	d. Return to Procedure and Step in effect and implement CSPs as necessary	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- When power is restored to any AC safeguards bus, recovery actions should continue starting with Step 70.
- If an SI signal exists or if an SI signal is actuated during this procedure, it should be reset to permit manual loading of equipment on AC safeguards buses.
- Service water pumps should remain available to automatically load on AC safeguards buses to provide EDG cooling.

47 Place Following Equipment Switches In Pull Out:

- Reactor coolant pumps:
  - 1P-1A
  - 1P-1B
- Charging pumps:
  - 1P-2A
  - 1P-2B
  - 1P-2C
- RHR pumps:
  - 1P-10A
  - 1P-10B
- Component cooling pumps:
  - 1P-11A
  - 1P-11B
- Motor-driven AFW pump:
  - 1P-53
- Main feed pumps:
  - 1P-28A
  - 1P-28B
- Heater drain tank pumps:
  - 1P-27A
  - 1P-27C
  - 1P-27B

(Step 47, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 47. continued from previous page)

- NON-running condensate pumps:
  - o 1P-25A
  - o 1P-25B
- NON-running circulating water pumps:
  - o 1P-30A
  - o 1P-30B
- Containment accident fans:
  - 1W-1A1
  - 1W-1B1
  - 1W-1C1
  - 1W-1D1
- Containment spray pumps:
  - 1P-14A
  - 1P-14B
- SI pumps:
  - 1P-15A
  - 1P-15B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Refer to ATTACHMENT J for a list of equipment that should be available during a loss of all AC power.

48 **Try To Restore Power To Any 480 Vac Safeguards Bus While Continuing With This Procedure**

- o Locally start G01 per ATTACHMENT A
- o Locally start G02 per ATTACHMENT B
- o Locally start G03 per ATTACHMENT C
- o Locally start G04 per ATTACHMENT D
- o Locally start G05 per ATTACHMENT E
- o Backfeed 480 Vac buses per ATTACHMENT F

RESPONSE NOT OBTAINED: Align equipment to alternate power source per ATTACHMENT G, ALIGNING EQUIPMENT TO ALTERNATE POWER SOURCE, while continuing with this procedure.

49 **Isolate RCP Seals:**

a. Shut at least one RCP seal return containment isolation valve:

- o 1CV-313A
- o 1CV-313

RESPONSE NOT OBTAINED: a. Locally shut RCP seal water return isolation valve outside containment:

- 1CV-313

b. Locally shut the following valves outside containment:

- RCP seal injection throttle valves:
  - 1CV-300A
  - 1CV-300B
- RCP component cooling return isolation valves:
  - 1CC-759A
  - 1CC-759B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

ELAP is short for Extended Loss of AC Power. It is longer than SBO coping time.

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50	**Check If AC Power Can Be Restored - WITHIN ONE HOUR**	**IF** AC power can **NOT** be restored in one hour, **THEN** perform the following while continuing on with this procedure: • FSG-5, INITIAL ASSESSMENT AND FLEX EQUIPMENT STAGING • FSG-4, ELAP DC BUS LOAD SHED/MANAGEMENT

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51	**Check CST Isolated From Condenser Hotwell:**	
	a. Ensure condenser hotwell low flow make-up valve - SHUT • 1CS-2125	a. **IF** valve **CANNOT** be manually shut, **THEN** locally shut upstream isolation valve: • 1CS-92
	b. Ensure condenser manual fill valve - SHUT • 1CS-86	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
52	**Check S/G Status:**	
	a. MSIVs - BOTH SHUT • 1MS-2018 • 1MS-2017	a. Manually shut MSIVs. **IF** any MSIVs **CANNOT** be shut, **THEN** locally shut MSIV(s) per ATTACHMENT H.
	b. MSIV bypass valves - BOTH SHUT • 1MS-234 • 1MS-236	b. Locally shut valves.
	c. Feedwater isolation valves - BOTH SHUT • 1CS-3124 • 1CS-3125	c. Perform the following: 1) Shut feedwater regulating valves: • 1FIC-466A • 1FIC-476A 2) Shut feedwater regulating bypass valves: • 1CS-480 • 1CS-481
	d. Blowdown isolation valves - SHUT • 1MS-5958 • 1MS-5959	d. Shut blowdown header isolation valves: • 1MS-2042 • 1MS-2045
53	**Check S/G's Available For Cooldown:**	
	a. Both S/G ADV's are: o Accessible o Capable of operating	a. **IF** an asymmetrical cooldown is required, **THEN** go to ATTACHMENT M, Asymmetrical Cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- **A faulted or ruptured S/G that is isolated should remain isolated.**
- **Alternate AFW pump suction supply will be necessary if CST level lowers to less than 4 ft.**

NOTE

If an ELAP is in progress and CST level lowers to 15.75 ft., FSG-6, ALTERNATE CST MAKEUP should be used if CSTs are available

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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
54	**Check Intact S/G Levels:**	
	a. S/G levels - GREATER THAN [51%] 32%	a. Maintain maximum AFW flow until level is greater than [51%] 32% in at least one S/G. **IF** an ELAP is **NOT** in progress, **THEN** perform ATTACHMENT K, ESTABLISHING HEAT SINK. **IF** an ELAP is in progress **AND** CST is NOT available, **THEN** perform FSG-2, ALTERNATE AFW SUCTION SOURCE **IF** an ELAP is in progress **AND** an alternate low pressure feedwater source is required due to 1P-29 unavailability, **THEN** perform the following: 1. **IF** 2P-29 **AND** associated suction source is available, **THEN** perform FSG-15 UNIT 1, CROSS TIE AFW. 2. **IF** 2P-29 AND associated suction source **NOT** available, **THEN** establish low pressure feedwater per ATTACHMENT L, ESTABLISHING LOW PRESSURE FEEDWATER FLOW.

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(Step 54, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 54. continued from previous page)

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b. Control AFW flow to maintain S/G levels between [51%] 32% and 63%

b. **IF** level in any S/G continues to rise in an uncontrolled manner, **THEN** isolate ruptured S/G(s):

1) Reset Loss Of Feedwater Turbine Trip

2) SHUT MDAFW pump discharge valves:

- o 1AF-4074A for S/G A
- o 1AF-4074B for S/G B

3) Shut turbine driven AFW pump discharge valves:

- o 1AF-4001 for S/G A
- o 1AF-4000 for S/G B

4) Shut steam supply to turbine-driven AFW pump:

- o 1MS-2020 for S/G A
- o 1MS-2019 for S/G B

5) Adjust ruptured S/G(s) atmospheric steam dump controller to 1050 psig

- o 1HC-468 for S/G A
- o 1HC-478 for S/G B

6) **WHEN** ruptured S/G(s) pressure is less than 1050 psig, **THEN** ensure atmospheric steam dump is shut:

- o 1MS-2016 for S/G A
- o 1MS-2015 for S/G B

IF atmospheric steam dump **CANNOT** be shut, **THEN** locally isolate atmospheric steam dump:

- o 1MS-227 for S/G A
- o 1MS-244 for S/G B

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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Steam supply to the turbine-driven AFW pump must be maintained from at least one S/G.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
55	**Check If S/G are NOT faulted:** • NO S/G pressure lowering in an uncontrolled manner **AND** • NO S/G completely depressurized	Isolate faulted S/G(s): a. Reset Loss Of Feedwater Turbine Trip. b. Shut motor driven AFW pump discharge valves: o 1AF-4074A for S/G A o 1AF-4074B for S/G B c. Shut turbine driven AFW pump discharge valves: o 1AF-4001 for S/G A o 1AF-4000 for S/G B d. Shut steam supply to turbine-driven AFW pump: o 1MS-2020 for S/G A o 1MS-2019 for S/G B e. Ensure atmospheric steam dump is shut: o 1MS-2016 for S/G A o 1MS-2015 for S/G B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
56	**Check If S/G Tubes Are <u>NOT</u> ruptured:**	Perform the following:
	a. Secondary system radiation monitor levels - NORMAL • Condenser air ejector: • 1RE-215 (3-5) • RE-225 (7-1) • S/G blowdown: • 1RE-219 (5-3) • 1RE-222 (1-7) • Main steam line: • 1RE-231 (3-9) for S/G A • 1RE-232 (5-2) for S/G B	1. Try to identify ruptured S/G(s) while continuing with this procedure. 2. **<u>WHEN</u>** ruptured S/G(s) identified, **<u>THEN</u>** isolate ruptured S/G(s): a) Reset Loss Of Feedwater Turbine Trip. b) Shut motor driven AFW pump discharge valves: o 1AF-4074A for S/G A o 1AF-4074B for S/G B c) Shut turbine driven AFW pump discharge valves: o 1AF-4001 for S/G A o 1AF-4000 for S/G B
	b. Request Chemistry to periodically sample both S/Gs for activity c. Request local surveys of main steam lines	d) Shut steam supply to turbine-driven AFW pump: o 1MS-2020 for S/G A o 1MS-2019 for S/G B
	b. Secondary activity samples and surveys - NORMAL (WHEN AVAILABLE)	e) Adjust ruptured S/G(s) atmospheric steam dump controller to 1050 psig: o 1HC-468 for S/G A o 1HC-478 for S/G B f) **<u>WHEN</u>** ruptured S/G(s) pressure is less than 1050 psig, **<u>THEN</u>** ensure atmospheric steam dump is shut: o 1MS-2016 for S/G A o 1MS-2015 for S/G B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
57	**Check DC Bus Loads:**	
	a. Check ELAP - IN PROGRESS	a. Go to Step 58.
	b. Check vital instrumentation - AVAILABLE • Any DC Bus Voltage GREATER THAN 107.5 Vdc • Vital instruments - REQUIRED INSTRUMENTS AVAILABLE o Red Instrument Bus Powered o White Instrument Bus Powered	b. Perform FSG-7 Unit 1, LOSS OF VITAL INSTRUMENTATION OR CONTROL POWER while continuing on with this procedure.
58	**Check CST Level**	
	a. CST Level - GREATER THAN 15.75 ft.	a. **IF** an ELAP is in progress **AND** CST is Available, **THEN** perform FSG-6, ALTERNATE CST MAKEUP while continuing with this procedure
	b. CST Level - GREATER THAN 4 ft.	b. **IF** an ELAP **NOT** in progress, **THEN** switch to alternate AFW suction supply per AOP-23, UNIT 1, ESTABLISHING ALTERNATE AFW SUCTION SUPPLY while continuing with this procedure. **IF** an ELAP in progress, **THEN** perform FSG-2, ALTERNATE AFW SUCTION SOURCE while continuing with this procedure

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

RCS temperature should be stabilized prior to taking action based on Low PZR level. This may require controlling steam generator feed rate and allowing temperature to return to 547°F.

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59 Monitor RCS Integrity

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Monitor RCS inventory – MAINTAINED AS EXPECTED • RCS subcooling based on CET's – Greater than [74°F] 35°F • PZR Level – Greater than [32%] 13%	a. Go to Step 60.
b. Check plant conditions – ELAP IN PROGRESS	b. Perform the following: 1) Consult with TSC to determine when plant cooldown should be initiated. 2) **WHEN** plant cooldown is desired, **THEN** go to Step 60. Continue with Step 62.
c. Check S/G ADVs – REMOTE CONTOL AVAILABLE	c. **WHEN** RCS cooldown is desired **AND** personnel are available to manually operate the ADVs, **THEN** go to Step 60.

****

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- S/G pressures should be maintained greater than 280 psig to prevent injection of accumulator nitrogen into the RCS.
- S/G level should be maintained greater than [51%] 32% in at least one intact S/G. If level cannot be maintained, S/G depressurization should be stopped until level is restored in at least one S/G.

NOTES

- CST level of 4 ft. is based on the decay heat load for 1 hour following a reactor trip without cool down of the RCS. Depressurization should not be started without either an adequate CST level, an alternate AFW suction supply or fire water aligned to CST.
- The S/G's should be depressurized a rate sufficient to maintain a cooldown rate in the RCS cold legs near 100°F/hr. This will minimize RCS inventory loss while cooling the RCP seals in a controlled manner.
- PZR level may be lost and reactor vessel upper head voiding may occur due to depressurization of S/Gs. Depressurization should not be stopped to prevent these occurrences.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
60	**Depressurize Intact S/G's To 320 psig:**	
	a. S/G levels - GREATER THAN [51%] 32% IN AT LEAST ONE S/G	a. Perform the following: 1) Maintain maximum AFW flow until level is greater than [51%] 32% in at least one S/G. 2) <u>IF</u> an ELAP is <u>NOT</u> in progress, <u>THEN</u> perform <u>ATTACHMENT K, ESTABLISHING HEAT SINK.</u>

(Step 60, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	(Step 60. continued from previous page)	
		3) **IF** an ELAP is in progress **AND** an alternate low pressure feedwater source is required due to 1P-29 unavailability, **THEN** perform the following: a) **IF** 2P-29 **AND** associated suction source is available, **THEN** perform FSG-15 UNIT 1, CROSS TIE AFW b) **IF** 2P-29 **AND** associated suction source **NOT** available, **THEN** establish low pressure feedwater per ATTACHMENT L, ESTABLISHING LOW PRESSURE FEEDWATER FLOW
		4) **WHEN** level is greater than [51%] 32% in at least one S/G, **THEN** do Steps 60.b through 60.e Continue with Step 61.
	b. Manually dump steam using S/G ADVs to maintain cooldown rate in RCS cold legs - LESS THAN 100°F/hr. o 1MS-2016 for S/G A o 1MS-2015 for S/G B	b. Locally dump steam using atmospheric steam dump. o 1MS-2016 for S/G A o 1MS-2015 for S/G B
	c. RCS cold leg temperatures - GREATER THAN 340°F	c. Perform the following 1) Control atmospheric steam dump to stop S/G depressurization. 2) Go to Step 61.
	d. S/G pressures - LESS THAN 320 psig.	d. **WHEN** S/G pressures lower to less than 320 psig, **THEN** do Step 60.e. Continue with Step 61.

(Step 60, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	(Step 60. continued from previous page)	
	e. Manually control atmospheric steam dumps to maintain S/G pressures at 320 psig: • 1MS-2016 for S/G A • 1MS-2015 for S/G B	e. Locally control atmospheric steam dumps to maintain S/G pressures at 320 psig. • 1MS-2016 for S/G A • 1MS-2015 for S/G B IF S/G pressure decreases to less than 320 psig without operation of atmospheric steam dumps or S/G feed, THEN perform FSG-9 UNIT 1 LOW DECAY HEAT TEMPERATURE CONTROL.

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61	**Check Reactor Subcritical:** a. Intermediate range channels-ZERO OR NEGATIVE STARTUP RATE • [1N-40B] • 1N-35 • 1N-36 b. Source range channels - ZERO OR NEGATIVE STARTUP RATE • [1N-40D] • 1N-31 • 1N-32	Control atmospheric steam dump to stop S/G depressurization and allow RCS to heat up: • 1MS-2016 for S/G A • 1MS-2015 for S/G B

****

NOTE

Depressurization of S/G's will result in SI actuation. SI should be reset to permit manual loading of equipment on AC safeguards bus.

62	**Check SI - ACTUATED**	WHEN SI is actuated, THEN do Steps 63, 64 and 65. Continue with Step 66.
63	**Reset SI**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
64	**Verify Containment Isolation:**	
	a. Containment isolation panels "A" and "B" - ALL LIGHTS LIT	a. Perform the following: 1) Manually actuate Containment Isolation. 2) **IF** any containment isolation valve is open, **THEN** manually shut valve(s). Refer to ATTACHMENT I, CONTAINMENT ISOLATION VALVES. **IF** any valve **CANNOT** be shut, **THEN** locally shut valve(s).
65	**Check Containment Pressure Recorder - HAS REMAINED LESS THAN 25 psig** • 1PR-968 • 1PR-969	Perform the following: a. Check containment spray actuated: • Annunciator C01 B 2-6, CONTAINMENT SPRAY, lit **IF** containment spray has **NOT** actuated, **THEN** manually actuate containment spray. b. Reset containment spray signal.
66	**Check Core Exit Thermocouple Temperatures - LESS THAN 1200°F** • 1TR-00001A • 1TR-00001B	**IF** core exit temperatures are greater than 1200°F **AND** rising, **THEN** go to SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDANCE INITIAL RESPONSE.
67	**Check Plant Conditions** • ELAP in progress • S/G depressurization to 320 psig -HAS BEEN COMPLETED	Go to Step 70.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
68	**Depressurize Intact S/G's For Maximum Passive Injection:**	
	a. Check FSG-8 UNIT 1, ALTERNATE RCS BORATION STATUS • COMPLETE • MAXIMUM PASSIVE INJECTION DIRECTED	a. **IF** required boration has been completed per FSG-8 UNIT 1, **THEN** go to Step 69. **IF NOT**, **THEN** Go to Step 70.
	b. Check the following: • Accumulator pressure indication available • FLEX PDG is in service per FSG-5, INITIAL ASSESMENT AND FLEX EQUIPMENT STAGING	b. Perform the following: 1) **WHEN** indication and FLEX Equipment is in place, **THEN** continue with Step 68.c through 68.j. 2) Continue with Step 70.

(Step 68, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 68. continued from previous page)

c. Check S/G narrow range levels greater than [51%] 32%

c. Perform the following:

1) Maintain maximum AFW flow until narrow range level greater then [51%] 32% in at least one S/G

2) **IF** an ELAP is in progress **AND** an alternate low pressure feedwater source is required due to 1P-29 unavailability, **THEN** perform the following:

a) **IF** 2P-29 **AND** associated suction source is available, **THEN** perform FSG-15 UNIT 1, CROSS TIE AFW.

b) **IF** 2P-29 **AND** associated suction source **NOT** available, **THEN** establish low pressure feedwater per ATTACHMENT L, ESTABLISHING LOW PRESSURE FEEDWATER FLOW.

3) **WHEN** narrow range level is greater than [51%] 32% in at least one S/G, **THEN** continue with Step 68.d through 68.j

4) Continue with Step 70.

d. Manually dump steam using S/G atmospheric dump valves to maintain cooldown rate in RCS cold legs - LESS THAN 100°F/hr

d. Locally dump steam using S/G atmospheric dump valves:

- 1MS-2016
- 1MS-2015

e. Monitor reactor - SUBCRITICAL

- Intermediate range channels - ZERO OR NEGATIVE START UP RATE
- Source range channels - ZERO OR NEGATIVE START UP RATE

e. Control S/G atmospheric dump valves to stop S/G depressurization and allow RCS to heat up

Go to Step 68.g.

(Step 68, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 68. continued from previous page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	f. Check either of the following conditions - MET: o S/G pressures - AT OR LESS THAN 120 psig o Accumulator pressures - AT OR LESS THAN 320 psig	f. Go to Step 68.d.
	g. Stabilize S/G pressure	
	h. Check accumulator pressure - AT OR LESS THAN 320 PSIG	h. Perform the following: 1) IF RCS subcooling is less than [74°F] 35°F, THEN go to Step 68.i. 2) Open RCS head vent valves: • 1RC-570A • 1RC-570B 3) **WHEN** accumulator pressure is less than 320 psig **OR** RCS subcooling is less than [74°F] 35°F, **THEN** perform the following: a) Close RCS head vent valves • 1RC-570A • 1RC-570B b) Go to Step 68.i. 4) Continue with Step 70.
	i. Check accumulator pressure stable	i. **IF** accumulator pressure continues to lower, **THEN** control S/G atmospheric dump valves to increase S/G pressure to 280 psig: o 1MS-2016 o 1MS-2015
	j. Isolate accumulators using FSG-10 UNIT 1, UNIT 1 PASSIVE RCS INJECTION ISOLATION	j. Control S/G atmospheric dump valves to lower S/G pressure to 320 psig o 1MS-2016 o 1MS-2015

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
69	**Depressurize intact S/G's to 150 psig for long term cooling:**	
	a. Check required boration per FSG-8 UNIT 1, ALTERNATE RCS BORATION - COMPLETE	a. Perform the following: 1) **WHEN** required boration per FSG-8 UNIT 1, is complete, **THEN** do Steps 69.b through 69.g. 2) Continue with Step 70.
	b. Perform FSG-10 UNIT 1, PASSIVE RCS INJECTION ISOLATION	
	c. Verify accumulators - ISOLATED • 1SI-841A • 1SI-841B	c. Perform FSG-10 UNIT 1, PASSIVE RCS INJECTION ISOLATION 1) **WHEN** accumulators are isolated, **THEN** continue with Step 69.d through 69.g 2) Continue to Step 70.
	d. Check S/G narrow range levels greater than [51%] 32% in at least one S/G	d. Perform the following: 1) Maintain maximum AFW flow until narrow range level greater than [51%] 32% in at least one S/G 2) **IF** an ELAP is in progress **AND** an alternate low pressure feedwater source is required due to 1P-29 unavailability, **THEN** perform the following: a) **IF** 2P-29 **AND** associated suction source is available, **THEN** perform FSG-15 UNIT 1, CROSS TIE AFW. b) **IF** 2P-29 **AND** associated suction source **NOT** available, **THEN** establish low pressure feedwater per ATTACHMENT L, ESTABLISHING LOW PRESSURE FEEDWATER FLOW. 3) **WHEN** narrow range level is greater than [51%] 32% in at least one S/G, **THEN** continue with Step 69.e through 69.g.

(Step 69, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 69. continued from previous page)

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
e.	Manually dump steam using S/G ADVs to maintain cooldown rate in RCS cold legs less than 100°F/hr. • 1MS-2016 • 1MS-2015	e. Locally dump steam using S/G ADVs: • 1MS-2016 • 1MS-2015
f.	Check S/G pressures - LESS THAN 150 psig	f. Perform the following: 4) **WHEN** S/G pressures lower to less than 150 psig, **THEN** go to Step 69.e. 5) Continue to Step 70.
g.	Manually control S/G ADVs at 150 psig: • 1MS-2016 • 1MS-2015	g. Locally control S/G ADVs to maintain S/G pressure at 150 psig **IF** S/G pressure can **NOT** be maintained at 150 psig, **THEN** perform FSG-9 UNIT 1, LOW DECAY HEAT TEMPERATURE CONTROL to stop the uncontrolled cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
70	**Check If AC Safeguards Power Is Restored:**	
	a. AC safeguards buses - AT LEAST ONE TRAIN ENERGIZED o 1A-05 and 1B-03 **OR** o 1A-06 and 1B-04	a. Continue to control RCS conditions and monitor plant status: 1) Check status of local actions: • AC power restoration (Step 48) • RCP seal isolation (Step 49) • Other local actions if applicable: o Gas Turbine (ATTACHMENT E) o AFW pump room cooling (Step 46 RNO 1)) o Condenser hotwell (Step 51) o MSIV and bypass valve closure (Step 52) o Faulted S/G isolation (Step 55) o Ruptured S/G isolation (Step 56) o CST makeup (Step 58) o Containment isolation (Step 64) 2) Periodically check status of spent fuel cooling: • Spent fuel pool level greater than 62' 8" El. o LI-40A o LI-40B o Installed Wall Tape • Spent fuel pool temperature less than 120°F. • **IF** level less than 62' 8" **OR** temperature greater than 120°F, **THEN** dispatch personnel to initiate actions per AOP-8F, LOSS OF SPENT FUEL POOL COOLING.

(Step 70, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 70. continued from previous page)

3) **IF** TDAFW Pump is operating, **THEN** gag OPEN Mini-recirc Valve:

- 1AF-4002

4) **IF** ELAP is in progress, **THEN** implement the following strategies as needed.

   o RCS Inventory Control

   IF pressurizer level is less than [44%] 24% and time and personnel are available, **OR** NR RVLIS is less than 25 ft. **AND** indication of reflux cooling, **THEN** perform FSG-1 UNIT 1, LONG TERM RCS INVENTORY CONTROL.

   o Boration

   IF time since event initiation is greater than 24 hours, **THEN** perform FSG-8 UNIT 1, ALTERNATE RCS BORATION.

   o Containment Cooling

   IF containment temperature greater than 175°F **OR** containment pressure greater than 48 psig, **THEN** perform FSG 12 UNIT 1, ALTERNATE CONTAINMENT COOLING.

5) OBSERVE CAUTION PRIOR TO STEP 54 and return to Step 54.

b. Check if any FSGs implemented.

b. Go to Step 71.

c. Perform FSG-13 UNIT 1, TRANSITION FROM FLEX EQUIPMENT.

d. Observe considerations specified in FSG-13 UNIT 1, TRANSITION FROM FLEX EQUIPMENT before starting equipment in subsequent steps and procedures.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

The loads placed on the energized AC safeguards bus should not exceed the capacity of the power source. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT, for KW ratings.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
71	**Verify Service Water System Operation:**	
	a. Service water header pressure - GREATER THAN OR EQUAL TO 50 psig	a. Manually start pumps and align valves as necessary to establish service water header pressure greater than or equal to 50 psig.
72	**Restore Battery Chargers:**	
	a. Trip and close contactor(s) for tripped battery chargers aligned to supply DC buses: o D-07 o D-08 o D-09 o D-107 o D-108 o D-109	
	b. Battery charger(s) - OPERATING	b. Restart affected battery charger(s) per the following: o 0-SOP-DC-001, 125 VDC SYSTEM, BUS D-01 & COMPONENTS o 0-SOP-DC-002, 125 VDC SYSTEM, BUS D-02 & COMPONENTS o 0-SOP-DC-003, 125 VDC SYSTEM, BUS D-03 & COMPONENTS o 0-SOP-DC-004, 125 VDC SYSTEM, BUS D-04 & COMPONENTS o 0-SOP-DC-005, 125 VDC SYSTEM, SWING BUSES & COMPONENTS o 0-SOP-DC-006, 125 VDC SYSTEM, NON-VITAL BUSES & COMPONENTS

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
73	**Stabilize S/G Pressures:**	
	a. Manually control atmospheric steam dumps: • 1MS-2016 for S/G A • 1MS-2015 for S/G B	a. Locally control atmospheric steam dumps. • 1MS-2016 for S/G A • 1MS-2015 for S/G B
74	**Verify Following Equipment Loaded On AC Safeguards Bus:**	
	a. Emergency AC lighting and plant alarms: o 1B-42 o 1B-32	a. Energize 1B-42 or 1B-32.
	b. B-33 and B-43 - AT LEAST ONE ENERGIZED	b. Energize B-33 or B-43.
	c. Communications: 1) Gai-tronics 2) Radio communications	c. Perform one of the following: a) <u>IF</u> B-33 is energized, <u>THEN</u> position transfer switch to B-33: • B-50 b) <u>IF</u> B-33 is <u>NOT</u> energized, <u>THEN</u> position transfer switch to B-43: • B-50

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

If RCP seal cooling was previously isolated, further cooling of the RCP seals will be established by natural circulation cooldown as directed in subsequent procedures.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
75	**Select Recovery Procedure:**	
	a. Check RCS subcooling based on core exit thermocouples - GREATER THAN [74°F] 35°F	a. Go to ECA-0. 2 UNIT 1, LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED.
	b. Check PZR level - GREATER THAN [32%] 13%	b. Go to ECA-0. 2 UNIT 1, LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED.
	c. Check SI equipment - HAS **NOT** ACTUATED UPON AC POWER RESTORATION	c. Go to ECA-0. 2 UNIT 1, LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED.
	d. Go to ECA-0. 1 UNIT 1, LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A

(Page 1 of 5)

G-01 LOCAL MANUAL START

A1	**Check Green POWER ON Light - LIT** • Panel C-64A • Panel C-34	IF green light is NOT lit, THEN transfer control power to alternate source: a. At PAB 8' elevation South of Unit 2 charging pumps, direct PAB operator to shut switch D31-01. b. At C-78, shift to alternate power by swapping paired breakers: • For annunciators, open breaker 1 and close breaker 2. • For start circuit 1, open breaker 3 and close breaker 4. • For control power, open breaker 5 and close breaker 6. • For field flash, open breaker 7 and close breaker 8.
A2	**Check Overspeed Trip Alarms - CLEAR** • Panel C-64A • Panel C-34	Reset mechanical overspeed trip and alarms as follows: a. At C-64, place 43/G-01-ESS, G-01 EDG Engine Start Selector Switch in LOCAL START. b. Reset mechanical overspeed trip. c. At C-64, place 43/G-01-ESS, G-01 EDG Engine Start Selector Switch in AUTO START.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A

(Page 2 of 5)

G-01 LOCAL MANUAL START

A3	**Emergency Start G-01:**	
	a. At C-34A, place local/remote transfer switches to LOCAL: • 43/G-01-1, G-01 EDG Transfer Switch • 43/G-01-2, G-01 EDG Transfer Switch	
	b. At C-34A, start G-01 by depressing EMERGENCY START pushbutton	b. **IF** G-01 will **NOT** emergency start, **THEN** manually start G-01: 1) At C-64, place 43/G-01-ESS, G-01 EDG Engine Start Selector Switch in LOCAL START. 2) At C-64, depress and hold ENGINE START pushbutton until engine speed rises to idle. 3) At C-64, raise engine speed to 900 rpm by depressing IDLE RELEASE pushbutton. 4) **IF** G-01 **CANNOT** be started, **THEN** do not continue and inform Control Room of G-01 status.
A4	**At C-64, Check G-01 Speed - GREATER THAN OR EQUAL TO 900 RPM**	Perform the following: a. At C-64, place 43/G-01-GOV, G-01 Governor Mode Selector Switch to HYD. b. **IF** diesel speed **NOT** greater than or equal to 900 RPM, **THEN** at C-64, raise speed using hydraulic governor control switch.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A

(Page 3 of 5)

G-01 LOCAL MANUAL START

CAUTION

Rubber gloves with leather protectors (located in VSG Room cubicle 39) and personnel safety equipment are required to locally operate FFC relay.

A5 Contact Control Room To Check G-01 Frequency - BETWEEN 59.7 Hz AND 60.3 Hz

Perform the following:

a. **IF** field is **NOT** flashed, **THEN** in C-34, manually actuate relay FFC for 3 seconds by pushing yellow tab on relay actuator located behind terminal connections.

b. At C-64, ensure 43/G-01-GOV, G01 EDG Mode Selector Switch in HYD, **THEN** adjust frequency using the hydraulic governor control switch.

c. **IF** hydraulic governor control switch **NOT** functional, **THEN** adjust frequency using SPEED control knob on faceplate of Woodward governor.

d. **IF** frequency **CANNOT** be maintained, **THEN** locally shutdown G-01:

1) At C-64, push both engine stop pushbuttons.

2) Pull fuel supply cut-off valve operator.

3) At C-34A, place output breakers in pull out:

- 1A52-60
- 2A52-73

4) Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A

(Page 4 of 5)

G-01 LOCAL MANUAL START

A6	Contact Control Room To Check G-01 Voltage - BETWEEN 4050 Vac AND 4300 Vac	Perform the following: a. <u>IF</u> field is <u>NOT</u> flashed, <u>THEN</u> in C-34, manually actuate relay FFC for 3 seconds by pushing yellow tab on relay actuator located behind terminal connections. b. Contact Control Room to maintain voltage between 4050 Vac and 4300 Vac by adjusting diesel loading. c. <u>IF</u> voltage <u>CANNOT</u> be maintained, <u>THEN</u> locally shutdown G-01: 1) At C-64, push both engine stop pushbuttons. 2) Pull fuel supply cut-off valve operator. 3) At C-34A, place output breakers in pull out: • 1A52-60 • 2A52-73 4) Return to <u>procedure and step in effect</u>.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A

(Page 5 of 5)

G-01 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A7	**Energize Bus 1A-05 From Normal Diesel Supply G-01:**	
	a. Check G-01 - RUNNING	a. Return to procedure and step in effect.
	b. In VSG room, ensure 1A-03 to 1A-05 bus tie breaker - OPEN • 1A52-57	b. Return to procedure and step in effect.
	c. At C-35A, ensure G-02 to 1A-05 bus tie breaker control switch - OPEN AND IN PULL OUT • 1A52-66	c. Return to procedure and step in effect.
	d. At C-34A, ensure G-01 to bus 1A-05 breaker control switch - IN AUTO • 1A52-60	d. Place G-01 to bus 1A-05 breaker control switch in AUTO.
	e. At C-34A, check G-01 to bus 1A-05 breaker - CLOSED • 1A52-60	e. Locally perform the following: 1) Try to auto-close breaker by placing control switch to trip position then release. 2) **IF** breaker will **NOT** auto-close, **THEN** manually close breaker control switch.
A8	**Check Bus 1B-03 - ENERGIZED**	**IF** bus 1A-05 energized, **THEN** energize bus 1B-03: a. Close bus 1A-05 feed to 1X-13: • 1A52-58, train A b. Close bus 1B-03 normal feed: • 1B52-16B, train A
A9	**Return To Procedure And Step In Effect**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B

(Page 1 of 5)

G-02 LOCAL MANUAL START

B1	**Dispatch Operator With Key Number 43 To G-02**	
B2	**Check Green POWER ON Light - LIT** • Panel C-65A • Panel C-35	IF green light is NOT lit, THEN transfer control power to alternate source: a. At PAB 8' elevation South of Unit 2 charging pumps, direct PAB watch to shut switch D41-01. b. At C-79, shift to alternate power by swapping paired breakers: • For annunciators, open breaker 1 and close breaker 2. • For start circuit 1, open breaker 3 and close breaker 4. • For control power, open breaker 5 and close breaker 6. • For field flash, open breaker 7 and close breaker 8.
B3	**Check Overspeed Trip Alarms - CLEAR** • Panel C-65A • Panel C-35	Reset mechanical overspeed trip and alarms as follows: a. At C-65, place 43/G-02-ESS, G-02 EDG Engine Start Selector Switch in LOCAL START. b. Reset mechanical overspeed trip. c. At C-65, place 43/G-02-ESS, G-02 EDG Engine Start Selector Switch in AUTO START.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B

(Page 2 of 5)

G-02 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B4	**Emergency Start G-02:**	
	a. At C-35A, place local/remote transfer switches to LOCAL: • 43/G-02-1, G-02 EDG Transfer Switch • 43/G-02-2, G-02 EDG Transfer Switch	
	b. At C-35A, start G-02 by depressing EMERGENCY START pushbutton	b. **IF** G-02 will **NOT** emergency start, **THEN** manually start G-02: 1) At C-65, place 43/G-02-ESS, G-02 EDG Engine Start Selector Switch in LOCAL START. 2) At C-65, depress and hold ENGINE START pushbutton until engine speed rises to idle. 3) At C-65, raise engine speed to 900 rpm by depressing IDLE RELEASE pushbutton. 4) **IF** G-02 **CANNOT** be started, **THEN** do not continue and inform Control Room of G-02 status.
B5	**At C-65, Check G-02 Speed - GREATER THAN OR EQUAL TO 900 RPM**	Perform the following: a. At C-65, place 43/G-02-GOV, G-02 Governor Mode Selector Switch to HYD. b. **IF** diesel speed **NOT** greater than or equal to 900 RPM, **THEN** at C-65, raise speed using hydraulic governor control switch.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B

(Page 3 of 5)

G-02 LOCAL MANUAL START

CAUTION

Rubber gloves with leather protectors (located in VSG Room cubicle 39) and personnel safety equipment are required to locally operate FFC relay.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B6	**Check G-02 Frequency - BETWEEN 59.7 Hz AND 60.3 Hz**	Perform the following: a. **IF** field is **NOT** flashed, **THEN** in C-35, manually actuate relay FFC for 3 seconds by pushing yellow tab on relay actuator located behind terminal connections. b. At C-65, ensure 43/G-02-GOV, G02 EDG Mode Selector Switch in HYD, **THEN** adjust frequency using the hydraulic governor control switch. c. **IF** hydraulic governor control switch **NOT** functional, **THEN** adjust frequency using SPEED control knob on faceplate of Woodward governor. d. **IF** frequency **CANNOT** be maintained, **THEN** locally shutdown G-02: 1) Push both engine stop pushbuttons. 2) Pull fuel supply cut-off valve operator. 3) At C-35A, place output breakers in pull out: • 1A52-66 • 2A52-67 4) Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B

(Page 4 of 5)

G-02 LOCAL MANUAL START

B7	**Check G-02 Voltage - BETWEEN 4050 Vac AND 4300 Vac**	Perform the following: a. IF field is NOT flashed, THEN in C-35, manually actuate relay FFC for 3 seconds by pushing yellow tab on relay actuator located behind terminal connections. b. Contact Control Room to maintain voltage between 4050 Vac and 4300 Vac by adjusting diesel loading. c. IF voltage CANNOT be maintained, THEN locally shutdown G-02: 1) Push both engine stop pushbuttons. 2) Pull fuel supply cut-off valve operator. 3) At C-35A, place output breakers in pull out: • 1A52-66 • 2A52-67 4) Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B

(Page 5 of 5)

G-02 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B8	**Energize Bus 1A-05 From Alternate Supply G-02:**	
	a. Check G-02 - RUNNING	a. Return to procedure and step in effect.
	b. In VSG Room, ensure 1A-03 to 1A-05 bus tie breaker - OPEN • 1A52-57	b. Return to procedure and step in effect.
	c. At C-34A, ensure G-01 to 1A-05 bus tie breaker control switch - OPEN AND IN PULL OUT • 1A52-60	c. Return to procedure and step in effect.
	d. At C-35A, unlock and place G-02 to bus 1A-05 breaker control switch - AUTO • 1A52-66	
	e. At C-35A, check G-02 to bus 1A-05 breaker - CLOSED • 1A52-66	e. Locally perform the following: 1) Try to auto-close breaker by placing control switch to trip position then release. 2) **IF** breaker will **NOT** auto-close, **THEN** manually close breaker control switch.
B9	**Check Bus 1B-03 - ENERGIZED**	**IF** bus 1A-05 energized, **THEN** energize bus 1B-03: a. Close bus 1A-05 feed to 1X-13: • 1A52-58, train A b. Close bus 1B-03 normal feed: • 1B52-16B, train A
B10	**Return To Procedure And Step In Effect**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C

(Page 1 of 6)

G-03 LOCAL MANUAL START

CAUTION

Flashing fields on both G-03 and G-04 at the same time will overload the 125 VDC power supply.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C1	**At D-28, Check D-28 125 VDC Control Power - GREATER THAN 115 Vdc** • BUS D-28 Voltage Selector Switch selected to D-28	Switch to alternate 125 VDC control power: a. **IF** power to D-40 **NOT** available, **THEN** return to procedure and step in effect. b. At D-40, place fused disconnect to ON: • D72-40-13 c. At D-28, place main power transfer switch to OFF: • D72-28-M d. At D-28, place alternate power transfer switch to ON: • D72-28-A e. At D-28, check voltage - GREATER THAN 115 Vdc. f. **IF** 125 VDC control power is **NOT** available, **THEN** return to procedure and step in effect.
C2	**At Panel C-101, Check Overspeed Trip Alarm - CLEAR**	Reset overspeed trip and alarm: a. Pull down on gray reset lever to the latched position. b. At C-101, depress ALARM RESET pushbutton.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C

(Page 2 of 6)

G-03 LOCAL MANUAL START

C3 Auto Start G-03:

a. At C-81, place 43/G-03-LRS, G-03 EDG Local/Remote Start Selector Switch to LOCAL

b. At C-81, ensure 43/G-03-GOV, G-03 EDG Mode Selector Switch in AUTO

c. At C-81, depress SHUTDOWN RESET pushbutton

d. Check G-03 - RUNNING

RESPONSE NOT OBTAINED:

d. Fast start G-03:

1) At C-81, depress FAST START pushbutton.

2) **IF** G-03 did **NOT** fast start, **THEN** manually start G-03:

a) At C-81, depress SHUTDOWN RESET pushbutton.

b) At C-81, depress VOLTAGE SHUTDOWN RESET pushbutton.

c) At C-81, depress ALARM RESET pushbutton.

d) At C-81, place 43/G-03-GOV G-03 EDG GOVERNOR MODE SELECTOR SWITCH, to HYD.

e) **IF** G-03 is **NOT** running, **THEN** depress FAST START pushbutton.

3) **IF** G-03 **CANNOT** be started, **THEN** return to procedure and step in effect.

C4 At C-81, Depress ALARM RESET Pushbutton

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C

(Page 3 of 6)

G-03 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C5	**At C-81, Check G-03 Frequency - BETWEEN 59.7 Hz AND 60.3 Hz**	Perform the following: a. **IF** field is **NOT** flashed, **THEN** at C-81, depress VOLTAGE SHUTDOWN RESET pushbutton. b. At C-81, ensure 43/G-03-GOV, G03 EDG Mode Selector Switch in HYD, **THEN** adjust frequency using the hydraulic governor control switch. c. **IF** hydraulic governor control switch **NOT** functional, **THEN** adjust frequency using SPEED control knob on faceplate of Woodward governor. d. **IF** frequency **CANNOT** be maintained, **THEN** locally shutdown G-03: 1) At C-81, push both engine stop pushbuttons: • Black engine stop • Red emergency stop 2) At C-81, place output breakers in pull out: • 1A52-80 • 2A52-87 3) Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C

(Page 4 of 6)

G-03 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C6	**At C-81, Check G-03 Voltage - BETWEEN 4050 Vac AND 4300 Vac** • AC Voltmeter - Generator	Perform the following: a. **IF** field is **NOT** flashed, **THEN** at C-81, depress VOLTAGE SHUTDOWN RESET pushbutton. b. **IF** voltage **CANNOT** be maintained, **THEN** locally shutdown G-03: 1) At C-81, push both engine stop pushbuttons: • Black engine stop • Red emergency stop 2) At C-81, place output breakers in pull out: • 1A52-80 • 2A52-87 3) Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C

(Page 5 of 6)

G-03 LOCAL MANUAL START

C7 **Energize Bus 1A-06 From Normal Diesel Supply G-03:**

a. Check G-03 - RUNNING

a. Return to procedure and step in effect.

b. In DGB room, ensure 1A-04 to 1A-06 bus tie breaker - OPEN

- 1A52-77

b. Return to procedure and step in effect.

c. At C-82, ensure G-04 to 1A-06 bus tie breaker control switch - OPEN AND IN PULL OUT

- 1A52-86

c. Return to procedure and step in effect.

d. At C-81, ensure G-03 to bus 1A-06 breaker control switch - IN AUTO

- 1A52-80

d. Place G-03 to bus 1A-06 breaker control switch in AUTO.

e. At C-81, check G-03 to bus 1A-06 breaker - CLOSED

- 1A52-80

e. Locally perform the following:

1) Try to auto-close breaker by placing control switch to trip position then release.

2) **IF** breaker will **NOT** auto-close, **THEN** perform the following:

a) Turn on synchronizing switch:

- G-03 Diesel Generator to bus 1A-06 Synchroscope

b) Manually close breaker control switch:

- 1A52-80

c) Turn off synchronizing switch:

- G-03 Diesel Generator to bus 1A-06 Synchroscope

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C

(Page 6 of 6)

G-03 LOCAL MANUAL START

C8	Check Bus 1B-04 - ENERGIZED	IF 1A-06 energized, THEN energize bus 1B-04: a. Close bus 1A-06 feed to 1X-14: • 1A52-84, train B b. Close bus 1B-04 normal feed: • 1B52-17B, train B
C9	Return To Procedure and step in effect	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D

(Page 1 of 6)

G-04 LOCAL MANUAL START

D1	**Dispatch Operator With Key Number 43 To G-04**	

CAUTION

Flashing fields on both G-03 and G-04 at the same time will overload the 125 VDC power supply.

D2	**At D-40, Check D-40 125 VDC Control Power - GREATER THAN 115 Vdc** • BUS D-40 Voltage Selector Switch selected to D-40	Switch to alternate 125 VDC control power: a. **IF** power to D-28 **NOT** available, **THEN** return to procedure and step in effect. b. At D-28, place fused disconnect to ON: • D72-28-13 c. At D-40, place main power transfer switch to OFF: • D72-40-M d. At D-40, place alternate power transfer switch to ON: • D72-40-A e. At D-40, check voltage - GREATER THAN 115 Vdc. f. **IF** 125 VDC control power is **NOT** available, **THEN** return to procedure and step in effect.
D3	**At Panel C-102, Check Overspeed Trip Alarm - CLEAR**	Reset overspeed trip and alarm: a. Pull down on gray reset lever to the latched position. b. At C-102, depress ALARM RESET pushbutton.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D
(Page 2 of 6)
G-04 LOCAL MANUAL START

D4 **Auto Start G-04:**

a. At C-82, place 43/G-04-LRS, G-04 EDG Local/Remote Start Selector Switch, to LOCAL

b. Ensure 43/G-04-GOV, G-04 EDG Governor Mode Selector Switch, in AUTO

c. At C-82, depress SHUTDOWN RESET pushbutton

d. Check G-04 - RUNNING

RESPONSE NOT OBTAINED:

d. Fast start G-04:

1) At C-82, depress FAST START pushbutton.

2) IF G-04 did NOT fast start, THEN manually start G-04:

a. At C-82, depress SHUTDOWN RESET pushbutton.

b. At C-82, depress VOLTAGE SHUTDOWN RESET pushbutton.

c. At C-82, depress ALARM RESET pushbutton.

d. At C-82, place 43/G-04-GOV, G-04 EDG Governor Mode Selector Switch, to HYD.

e. IF G-04 is NOT running, THEN depress FAST START pushbutton.

3) IF G-04 CANNOT be started, THEN return to procedure and step in effect.

D5 **At C-82, Depress ALARM RESET Pushbutton**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D

(Page 3 of 6)

G-04 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
D6	**At C-82, Check G-04 Frequency - BETWEEN 59. 7 HZ AND 60. 3 HZ**	Perform the following: a. <u>**IF**</u> field is <u>**NOT**</u> flashed, <u>**THEN**</u> at C-82, depress VOLTAGE SHUTDOWN RESET pushbutton. b. At C-82, ensure 43/G-04-GOV, G04 EDG Mode Selector Switch in HYD, <u>**THEN**</u> adjust frequency using the hydraulic governor control switch. c. <u>**IF**</u> hydraulic governor control switch <u>**NOT**</u> functional, <u>**THEN**</u> adjust frequency using SPEED control knob on faceplate of Woodward governor. d. <u>**IF**</u> frequency <u>**CANNOT**</u> be maintained, <u>**THEN**</u> locally shutdown G-04: 1) At C-82, push both engine stop pushbuttons: • Black engine stop • Red emergency stop 2) At C-82, place output breakers in pull out: • 1A52-86 • 2A52-93 3) Return to <u>procedure and step in effect</u>.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D

(Page 4 of 6)

G-04 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
D7	**At C-82, Check G-04 Voltage - BETWEEN 4050 Vac AND 4300 Vac** • AC Voltmeter - Generator	Perform the following: a. **IF** field is **NOT** flashed, **THEN** at C-82, depress VOLTAGE SHUTDOWN RESET pushbutton. b. **IF** voltage **CANNOT** be maintained, **THEN** locally shutdown G-03: 1) At C-82, push both engine stop pushbuttons: • Black engine stop • Red emergency stop 2) At C-82, place output breakers in pull out: • 1A52-86 • 2A52-93 3) Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D

(Page 5 of 6)

G-04 LOCAL MANUAL START

D8 Energize Bus 1A-06 From Alternate Supply G-04:

a. Check G-04 - RUNNING

a. Return to procedure and step in effect.

b. In DGB room, ensure 1A-04 to 1A-06 bus tie breaker - OPEN

- 1A52-77

b. Return to procedure and step in effect.

c. At C-81, ensure G-03 to 1A-06 bus tie breaker control switch - OPEN AND IN PULL OUT

- 1A52-80

c. Return to procedure and step in effect.

d. At C-82 unlock and place G-04 to bus 1A-06 breaker control switch in AUTO

- 1A52-86

e. At C-82, check G-04 to bus 1A-06 breaker - CLOSED

- 1A52-86

e. Locally perform the following:

1) Try to auto-close breaker by placing control switch to trip position then release.

2) **IF** breaker will **NOT** auto-close, **THEN** perform the following:

a) Turn on synchronizing switch:

- G-04 Diesel Generator to bus 1A-06 Synchroscope

b) Manually close breaker control switch:

- 1A52-86

c) Turn off synchronizing switch:

- G-04 Diesel Generator to bus 1A-06 Synchroscope

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D

(Page 6 of 6)

G-04 LOCAL MANUAL START

D9	Check Bus 1B-04 - ENERGIZED	IF 1A-06 energized, THEN energize bus 1B-04: a. Close bus 1A-06 feed to 1X-14: • 1A52-84, train B b. Close bus 1B-04 normal feed: • 1B52-17B, train B
D10	Return To Procedure and step in effect	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E

(Page 1 of 6)

POWER RESTORATION USING GAS TURBINE

E1 **Start G-05 Gas Turbine**

NOTE

When G-501, Auxiliary Diesel Generator is running, the G-501 fuel tank MUST be filled by an operator at 45 minute intervals. The estimated maximum G-501 run time is one hour.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	a. Check G-501 GT GENERATOR AUXILIARY DG RUNNING light - LIT	a. At G-501 Engine Control Panel, place Auto-Off-Start selector switch to Start position
	b. Check G-05 REMOTE STATION OPERATIVE light - LIT	b. Locally, Align G-501 output to supply G-05 Auxiliaries by performing the following: 1) Locally at B-507, place 43/52-N-52-E G-05 Auxiliaries Auto/Manual Transfer Switch to MAN. 2) Locally at 52-T, Emergency Pwr to B-502 TSC, place 52T switch to OPEN. 3) Locally at B-507, place 52-E Emer Pwr to G-05 GT Gen/Auxiliaries Control Switch to CLOSE.
	c. Check READY TO START light - LIT	c. Align GT Auxiliaries per OI 110, GAS TURBINE OPERATION
	d. Place GENERATOR RATE SELECTOR switch to NORMAL	
	e. Depress START pushbutton	e. Locally start G-05 per OI 110, GAS TURBINE OPERATION
	f. Depress MINIMUM LOAD pushbutton	
E2	**Check G-05 GT GENERATOR READY TO SYNCHRONIZE Light - LIT**	DO NOT CONTINUE until G-05 READY TO SYNCHRONIZE light is lit.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E

(Page 2 of 6)

POWER RESTORATION USING GAS TURBINE

E3	**Ensure POWER TO/FROM H-01 BUS Breaker - CLOSED** • H52-10	
E4	**Ensure The Following Electrical Breakers - OPEN** • 1A52-01, bus 1A-01 Normal Feed • 1A52-36, bus 1A-03 Normal Feed • 1A52-37, 1A-03 to 1A-01 Bus Tie Breaker • 1A52-40, 1A-03 to 2A-03 Bus Tie Breaker • 1A52-56, bus 1A-04 Normal Feed • 1A52-55, 1A-04 to 1A-02 Bus Tie Breaker • 1A52-17, bus 1A-02 Normal Feed • 1A52-52, 1A-04 to 2A-04 Bus Tie Breaker	
E5	**<u>IF</u> BOTH units have entered ECA-0.0, <u>THEN</u> isolate the 13.8 kV Bus from switchyard by ensuring the following breakers open:** • H52-20 • H52-30	Ensure the following breakers are open: • H52-20 • H52-31 • H52-21
E6	**Energize Bus H-01 And Bus H-02 From G-05:** a. Turn on G-05 GT GENERATOR MAIN BREAKER SYNCHROSCOPE switch b. Close G-05 main breaker: • H52-G05 c. Turn on H-02 TO H-01 BUS TIE SYNCHROSCOPE switch d. Close H-02 to H-01 bus tie breaker: • H52-21	<u>IF</u> breakers will <u>NOT</u> close, <u>THEN</u> perform the following: 1. Check status of the following lockouts: • 1X-04 lockout • H-01 bus lockout • H-02 bus lockout 2. <u>If</u> lockouts are <u>NOT</u> tripped, <u>THEN</u> locally shut stored energy breakers • H52-G05 • H52-21

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E

(Page 3 of 6)

POWER RESTORATION USING GAS TURBINE

E7	**Continue Operation Of Gas Turbine Per OI-110, GAS TURBINE OPERATION**	
E8	**Check Bus H-02 - ENERGIZED**	IF H-02 CANNOT be energized, THEN perform the following: a. Evaluate the plant status to determine if the gas turbine is required to support Appendix R or other plant equipment. b. Shut down gas turbine if desired. c. Operate G-501 to power the TSC per OI-35, ELECTRICAL EQUIPMENT OPERATION. d. Return to procedure and step in effect.

NOTE

Synchroscope operation required for 1A-03 / 1A-04 feed breaker closure.

E9	**Restore Power To Bus 1A-03 And Bus 1A-04:** a. Reset and close bus H-02 feed to 1X-04: • H52-22 b. Reset and close bus 1A-03 normal feed: • 1A52-36 c. Reset and close bus 1A-04 normal feed: • 1A52-56	
E10	**Check Bus 1A-03 And Bus 1A-04 - AT LEAST ONE ENERGIZED**	Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E

(Page 4 of 6)

POWER RESTORATION USING GAS TURBINE

E11	**Energize Bus 1A-05 From 1A-03:**	
	a. Check bus 1A-03 - ENERGIZED	a. Go to Step E12.
	b. Ensure G-01 to bus 1A-05 breaker - OPEN • 1A52-60	
	c. Ensure G-02 to bus 1A-05 breaker - OPEN • 1A52-66	
	d. Turn on synchronizing switch for 1A-03 to 1A-05 bus tie breaker: • 1A52-57	
	e. Trip and close 1A-03 to 1A-05 bus tie breaker: • 1A52-57	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E

(Page 5 of 6)

POWER RESTORATION USING GAS TURBINE

E12	**Energize Bus 1A-06 From 1A-04:**	
	a. Check bus 1A-04 - ENERGIZED	a. Go to Step E13.
	b. Ensure G-03 to bus 1A-06 breaker - OPEN • 1A52-80	
	c. Ensure G-04 to bus 1A-06 breaker - OPEN • 1A52-86	
	d. Trip and close 1A-04 normal feed to 1A-06: • 1A52-54	
	e. Turn on synchronizing switch for 1A-04 to 1A-06 bus tie breaker: • 1A52-77	
	f. Trip and close 1A-04 to 1A-06 bus tie breaker: • 1A52-77	
E13	**Check Bus 1A-05 And Bus 1A-06 - AT LEAST ONE ENERGIZED**	Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E

(Page 6 of 6)

POWER RESTORATION USING GAS TURBINE

E14	**Check 480 Vac Safeguards Buses - AT LEAST ONE ENERGIZED** • 1B-03, train A • 1B-04, train B	Try to restore power to either bus as follows: a. IF bus 1A-05 energized, THEN energize bus 1B-03: 1) Close bus 1A-05 feed to 1X-13: • 1A52-58, train A 2) Close bus 1B-03 normal feed: • 1B52-16B, train A b. IF 1A-06 energized, THEN energize bus 1B-04: 1) Close bus 1A-06 feed to 1X-14: • 1A52-84, train B 2) Close bus 1B-04 normal feed: • 1B52-17B, train B
E15	**Return To Step 46.**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT F

(Page 1 of 3)

BACKFEED TO 480 VAC SAFEGUARDS BUSES

F1	**Check 1A-01 And 1A-02 - At LEAST ONE ENERGIZED**	Perform the following: a. Energize 1A-01 and 1A-02 per AOP-18, ELECTRICAL SYSTEM MALFUNCTION.
F2	**Energize Bus 1B-01 From Bus 1A-01 And Bus 1B-02 From Bus 1A-02:** a. Close bus 1A-01 feed to 1X-11: • 1A52-02, train A b. Close bus 1B-01 normal feed: • 1B52-04B, train A c. Close bus 1A-02 feed to 1X-12: • 1A52-15, train B d. Close bus 1B-02 normal feed: • 1B52-05B, train B	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT F

(Page 2 of 3)

BACKFEED TO 480 VAC SAFEGUARDS BUSES

NOTE

The breaker trip for transformers 1X-11 and 1X-12 is 180 amps each. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT, for equipment load ratings.

F3 Backfeed Bus 1B-03 From Bus 1B-01 And Bus 1B-04 From Bus 1B-02:

a. Place Unit 1 service water pumps in pull out:

- P-32A, train A
- P-32B, train A
- P-32C, train B

b. Place DC control power fuse block for each tie breaker to OFF:

1) In 1B-03, place DC control power fuse block for 1B-01 to 1B-03 bus tie breaker to OFF:

   - 1B52-15C, train A

2) In 1B-04, place DC control power fuse block for 1B-04 to 1B-02 bus tie breaker to OFF:

   - 1B52-18C, train B

c. Ensure normal feed breakers open:

- 1B52-16B for 1B-03
- 1B52-17B for 1B-04

d. Locally close tie breakers:

- 1B52-15C, 1B-03 to 1B-01 bus tie breaker
- 1B52-18C, 1B-04 to 1B-02 bus tie breaker

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT F

(Page 3 of 3)

BACKFEED TO 480 VAC SAFEGUARDS BUSES

F4 Return To Procedure and step in effect

-END-

ATTACHMENT G

(Page 1 of 3)

ALIGNING EQUIPMENT TO ALTERNATE POWER SOURCE

G1 **IF power is NOT established to H-01, THEN return to procedure and step in effect.**

G2 **IF placing charging pump 1P-2A in operation from alternate power supply, THEN perform the following:**

a. Locally ensure 1P-2A normal feeder breaker open:

- 1B52-13A

b. Outside charging pump cubicles, align 1P-2A charging pump normal/alternate transfer switch to alternate power source:

- At 1B313A-B854B, place transfer switch in A2 ON - A4 OFF position.

c. Place 1P-2A charging pump controller in MANUAL at minimum speed:

- 1HC-428A

d. At C-45, Alternate Shutdown Control Panel, close 1P-2A alternate feeder breaker:

- B52-54B

e. At 1C04, place 1P-2A charging pump switch to START.

f. IF the pump fails to start, THEN wait for the VFD to initialize and then place the control switch to start.

g. Adjust charging line flow controller to full open:

- 1HC-142

h. At 1C04, place "In Alternate Control" placard next to 1P-2A charging pump control switch.

ATTACHMENT G

(Page 2 of 3)

ALIGNING EQUIPMENT TO ALTERNATE POWER SOURCE

CAUTION

If an undervoltage occurs on B-08 or B-09, local manual restart of associated service water pump is required.

G3 **IF placing a service water pump in operation from bus B-08, THEN perform the following:**

a. Select one service water pump to be powered from alternate supply:

- o P-32B
- o P-32F

b. Locally ensure normal feeder breaker for selected service water pump open:

- o 1B52-11C for P-32B
- o 2B52-34B for P-32F

c. In Room G-01, place alternate power selector switch to desired service water pump:

- B854D

d. In Room G-01, place normal/alternate transfer switch for selected service water pump to B-08 power supply:

- o 1B311C-B854D for P-32B
- o 2B334B-B854D for P-32F

e. At C-45, Alternate Shutdown Control Panel, check selected service water pump switches properly aligned to B-08 power supply using indicating lights.

f. At C-45, close P-32B/F alternate feeder breaker:

- B52-54D

g. At C01, place "In Alternate Control" placard next to service water pump control switch.

ATTACHMENT G

(Page 3 of 3)

ALIGNING EQUIPMENT TO ALTERNATE POWER SOURCE

G4 **IF placing a service water pump in operation from bus B-09, THEN perform the following:**

a. Select one service water pump to be powered from alternate supply:

- o P-32C
- o P-32E

b. Locally ensure normal feeder breaker for selected service water pump open:

- o 1B52-20C for P-32C
- o 2B52-27C for P-32E

c. In Room G-02, place alternate power selector switch to desired service water pump:

- B957D

d. In Room G-02, place normal/alternate transfer switch for selected service water pump to B-09 power supply:

- o 1B420C-B957D for P-32C
- o 2B427C-B957D for P-32E

e. At C-45, Alternate Shutdown Control Panel, check selected service water pump switches properly aligned to B-09 power supply using indicating lights.

f. At C-45, close P-32C/E alternate feeder breaker:

- B52-57D

IF service water pump P-32C/E breaker B52-57D failed to close or remain closed, **THEN** perform the following:

1) At B-09 place the B52-57D breaker charging spring motor control switch to ON.

2) **WHEN** the breaker closing spring is charged, **THEN** place the B52-57D breaker charging spring motor control switch to OFF.

3) Close service water pump P-32C/E breaker B52-57D.

g. At C01, place "In Alternate Control" placard next to service water pump control switch.

G5 **Return to procedure and step in effect.**

-END-

ATTACHMENT H

(Page 1 of 3)

LOCAL SHUTTING OF MSIV

NOTE

Throughout this attachment, "affected" refers to the MSIV being shut.

H1 IF Control Room annunciator C02 D 4-5, UNIT 1 SAFEGUARDS DC CONTROL POWER FAILURE, is lit, THEN ensure MSIV solenoid power supplies are on:

- D72-16-2, train A
- D72-21-2, train B

H2 Shut affected MSIV using local pushbutton:

a. IF shutting 1MS-2018, THEN press both pushbuttons in 1RK-33.

- 1MS PB-2018A, train A
- 1MS PB-2018B, train B

b. IF shutting 1MS-2017, THEN press both pushbuttons in 1RK-34.

- 1MS PB-2017A, train A
- 1MS PB-2017B, train B

ATTACHMENT H

(Page 2 of 3)

LOCAL SHUTTING OF MSIV

H3 IF affected MSIV will NOT shut using local pushbuttons, THEN vent off air from affected MSIV as follows:

a. Obtain two painted red combination wrenches. (MSIV cabinet)

b. Shut instrument air supply valve to affected MSIV.

- o IA-638 for 1MS-2018
- o IA-636 for 1MS-2017

c. Remove end cap downstream of S/G header test isolation valve of affected MSIV.

- o 1MS-331A for S/G A
- o 1MS-331B for S/G B

d. Vent air off affected MSIV operator by opening S/G header test isolation valve.

- o 1MS-331A for S/G A
- o 1MS-331B for S/G B

e. Return painted red combination wrenches to MSIV cabinet.

f. Red lock combination wrenches.

ATTACHMENT H

(Page 3 of 3)

LOCAL SHUTTING OF MSIV

CAUTION

Rapid movement of the ratcheting torque wrench may occur when valve movement starts.

NOTE

The Direction Position Switch on the ratchet is color coded and unit designated for unit applicability.

H4 IF affected MSIV will NOT shut by venting off air, THEN perform the following:

a. On 66' PAB near waste distillate tanks, open the M&TE box using the M&TE key and obtain the following equipment:

- One ratcheting torque wrench
- One 2 3/4" socket

b. Apply wrench to the nut on the end of the valve shaft and shut affected MSIV.

H5 Notify Control Room of affected MSIV status.

-END-

ATTACHMENT I

(Page 1 of 2)

CONTAINMENT ISOLATION VALVES

PANEL A		
COMMENT	DESCRIPTION	TRAIN
1CV-1296	Auxiliary charging line	A
1RC-538	Pressurizer relief tank to gas analyzer	A
1WG-1788	Reactor coolant drain tank to gas analyzer	A
1WL-1698	Reactor coolant drain tank to -19 ft sump	A
1WL-1003A	Reactor coolant drain tank pump suction	A
1WL-1003B	Reactor coolant drain tank pump suction	A
1RC-508	Reactor makeup water to containment	A or B
1RC-539	Pressurizer relief tank to gas analyzer	B
1WG-1789	Reactor coolant drain tank to gas analyzer	B
1SI-846	Accumulator nitrogen supply	A or B
1WL-1721	Reactor coolant drain tank pumps suction	B
1WL-1723	Sump A drain	A
1SC-951	Pressurizer steam sample	A
1SC-953	Pressurizer liquid sample	A
1WL-1728	Sump A drain	B
1SC-966A	Pressurizer steam sample	A or B
1SC-966B	Pressurizer liquid sample	A or B

ATTACHMENT I

(Page 2 of 2)

CONTAINMENT ISOLATION VALVES

PANEL B		
COMMENT	DESCRIPTION	TRAIN
1CC-769	Component cooling water outlet from excess letdown heat exchanger	A or B
1CV-313	Reactor coolant pump seal return	A
1CV-371	Letdown line	A
1MS-5958	Steam generator blowdown	A or B
1MS-5959	Steam generator blowdown	A or B
1WG-1786	Reactor coolant drain tank vent	A
1CV-313A	Reactor coolant pump seal return	B
1CV-371A	Letdown line	B
1WG-1787	Reactor coolant drain tank vent	B
1RM-3200C	RE-211/212 supply	A
1RM-3200A	RE-211/212 return	A or B
1MS-2083	Steam generator A sample	A or B
1MS-2084	Steam generator B sample	A or B
1SC-955	Reactor coolant hot leg sample	A
1IA-3047	Instrument air line	A or B
1RM-3200B	RE-211/212 supply	B
1SC-966C	Reactor coolant hot leg sample	A or B
1IA-3048	Instrument air line	A or B

-END-

ATTACHMENT J

(Page 1 of 1)

EQUIPMENT AVAILABLE DURING LOSS OF ALL AC POWER

COMMUNICATIONS

- NAWAS
- Outside line in TSC
- ATCo telephone
- Operations radio talkgroups OPS 1, OPS 2, OPS 3 and OPS 4
- Interplant trouble circuit
- Motorola radio to radio channels PNT to PNT1 and PNT to PNT 2
- Gai-tronics
- PBX telephone system
- NRC telephone
- Security radio talkgroups Security 1 and Security 2

LIGHTING

Station Battery Fixed (EB) DC Lights

- Vital switchgear room
- Diesel room (doorway only)
- Control Room
- Cable spreading room

Fixed Emergency DC Sealed Beam Lanterns (EL)

- These lanterns are located along all entry and egress routes to both safe and alternate shutdown equipment and total approximately 100.
- Each individual lantern battery pack is designed for up to 8 hours of operation allowing for restoration of normal AC power.

Portable Emergency Lanterns

- C59
- TSC
- Auxiliary feed tunnel
- Unit 2 non-nuclear room
- Brigade ready
- Fire cart, Unit 2 turbine hall, Elevation 8 ft.

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT K

(Page 1 of 2)

ESTABLISHING HEAT SINK

K1	**Verify AFW flow:**	
	a. AFW flow-GREATER THAN **OR** EQUAL TO 230 gpm	a. Establish feed flow from SSG pump as follows: 1) Ensure adequate power is available. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT, for KW ratings. 2) Place selected Stripping Logic Override Switch to the OVERRIDE position. • P-38BX-CS 3) Start Standby Steam Generator feed pump: • P-38B 4) Verify valve alignment: a) Ensure Unit 1 valve OPEN: • AF-4021, train B b) Ensure Unit 2 valve SHUT: • AF-4020 c) Manually align valve as necessary to establish flow greater than or equal to 230 gpm: • 1AF-4019 d) **IF** feed flow has been established, **THEN** return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT K

(Page 2 of 2)

ESTABLISHING HEAT SINK

IF any feed flow to at least one S/G is verified, THEN perform the following:

1) Maintain feed flow to restore S/G level to greater than [51%] 32%.

IF feed flow is NOT verified, THEN establish AFW from Unit 2 by performing the following:

2) Locally open the following valves:

- 1AF-192, Unit 1 AFW Cross-connect Valve
- 2AF-192, Unit 2 AFW Cross-connect Valve

3) Start the Unit 2 motor driven AFW pump and monitor total flow using Unit 2 indications:

- 2P-53

4) Align AFW valve(s) to provide GREATER THAN 230 gpm:

- 1AF-4074A, for S/G A
- 1AF-4074B, for S/G B

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT L
(Page 1 of 3)

ESTABLISHING LOW PRESSURE FEEDWATER FLOW

L1 Initiate FSG-3 UNIT 1, ALTERNATE LOW PRESSURE FEEDWATER.

L2 WHEN low pressure feedwater source is ready to provide flow, THEN continue with Step L3.

NOTE

If SI accumulators are isolated or vented and S/G pressures are less than or equal to 320 psig, then S/G pressures and core exit thermocouples should be controlled at less than or equal to existing values.

L3 IF shutoff pressure of low pressure feedwater source is greater than 320 psig, THEN perform the following:

a. Depressurize selected S/G(s) below shutoff pressure of low pressure source using S/G atmospheric steam dumps manually or locally.

- 1MS-2016
- 1MS-2015

b. IF the source range or intermediate range startup rate is positive, THEN perform the following:

1) Control RCS temperature using S/G ADVs to maintain the reactor subcritical.

   - 1MS-2016
   - 1MS-2015

2) WHEN S/G ADVs have no little effect on RCS temperature, THEN decrease S/G pressure below shutoff pressure of low pressure source.

c. Establish feed flow at target value based on decay heat.

- Reference FSG-3 UNIT 1, ALTERNATE LOW PRESSURE FEEDWATER ATTACHMENT A for valves.

d. Maintain S/G pressure at 320 psig using SG ADVs manually or locally:

- 1MS-2016
- 1MS-2015

e. IF the reactor remains subcritical, THEN control feed flow to maintain S/G narrow range level between [51%] 32% and 63%.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT L
(Page 2 of 3)
ESTABLISHING LOW PRESSURE FEEDWATER FLOW

f. IF the source range or intermediate range startup rate is positive, THEN perform the following:

- Control feed flow to increase core exit thermocouples between 428°F and 547°F to prevent criticality.
- WHEN boration or Xenon adds negative reactivity to allow cooldown to 428°F, THEN continue depressurization.

NOTE

In order to establish adequate feedwater flow in the following step, it will be necessary to feed and steam S/G's that are close to dry and at low pressure.

L4 **IF shutoff pressure of low pressure feedwater source is less than 320 psig, THEN perform the following:**

a. Depressurize selected S/G(s) below shutoff pressure of low pressure source using S/G atmospheric steam dumps manually or locally while maintaining core exit thermocouples between 428°F and 547°F.

- 1MS-2016
- 1MS-2015

b. IF the source range or intermediate range startup rate is positive, THEN perform the following:

1) Control RCS temperature using S/G ADVs to maintain the reactor subcritical.
   - 1MS-2016
   - 1MS-2015
2) WHEN S/G ADVs have no little effect on RCS temperature, THEN decrease S/G pressure below shutoff pressure of low pressure source.

c. Establish feed flow at target value based on decay heat.

- Reference FSG-3 UNIT 1, ALTERNATE LOW PRESSURE FEEDWATER ATTACHMENT A for valves.

d. Lower S/G pressure as required to increase feed flow.

e. IF the source range or intermediate range startup rate is positive, THEN perform the following:

- Control feed flow to increase core exit thermocouples between 428°F and 547°F to prevent criticality.
- WHEN boration or Xenon adds negative reactivity to allow cooldown to 428°F, THEN continue depressurization.

f. Control feed flow and S/G pressures to maintain core exit TCs at 428°F

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT L

(Page 3 of 3)

ESTABLISHING LOW PRESSURE FEEDWATER FLOW

L5 **IF the low pressure source maintains S/G narrow range level between [51%] 32% and 63%, THEN go to Step L9.**

L6 **IF another feedwater source is subsequently established to the same S/G(s), THEN control S/G ADVs steam dump to allow S/G pressure to increase and recover S/G level, while avoiding an uncontrolled cooldown.**

L7 **IF another feedwater source restores a different S/G narrow range level above [51%] 32% and feeding with the low pressure source is no longer necessary to remove decay heat, THEN perform the following:**

a. Isolate the low pressure source from the feed line.

b. IF S/G level fed from the low pressure source is less than [51%] 32%, THEN close associated S/G ADV.

L8 **WHEN ECA-0.0 is exited, THEN consider the following in subsequent recovery:**

- Do not enter CSP-H.1 due to low S/G levels and low feedwater flow alone, unless low pressure feed source is not adequate to maintain core exit TCs stable.
- IF main feedwater OR condensate is used, THEN aligning to different S/Gs may be desired to limit risk of water hammer in depressurized feed lines.
- IF another feedwater source is established to the same S/G(s), THEN control steam dump to allow S/G pressure to increase and recover S/G level, while avoiding an uncontrolled cooldown.

L9 **Return to FSG-3 UNIT 1, ALTERNATE LOW PRESSURE FEEDWATER Step 7.**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT M
(Page 1 of 7)
ASYMMETRICAL COOLDOWN

NOTES

- **Boration will take approximately 3 hours to complete when utilizing the BAST, approximately 6 hours for the RWST.**
- **Pressurizer level should be allowed to rise towards the end of the boration to support the cooldown**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
M1	**Initiate RCS Boration**	
	• OP-5B, Blender Operation / Dilution / Boration	
	• FSG-8 Unit 1, Alternate RCS Boration	
M2	**Check CST Level:**	
	a. CST Level - GREATER THAN 15.75 ft.	a. **IF** an ELAP is in progress **AND** CST is Available, **THEN** perform FSG-6, ALTERNATE CST MAKEUP while continuing with this procedure.
	b. CST Level - GREATER THAN 4 ft.	b. **IF** an ELAP **NOT** in progress, **THEN** switch to alternate AFW suction supply per AOP-23, UNIT 1, ESTABLISHING ALTERNATE AFW SUCTION SUPPLY while continuing with this procedure. **IF** an ELAP in progress, **THEN** perform FSG-2, ALTERNATE AFW SUCTION SOURCE while continuing with this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT M
(Page 2 of 7)
ASYMMETRICAL COOLDOWN

****

M3	**Check Intact S/G Level:**	
	a. S/G level - GREATER THAN [51%] 32%	a. Maintain maximum AFW flow until level is greater than [51%] 32% in at least one S/G 1) **IF** an ELAP is **NOT** in progress, **THEN** perform ATTACHMENT K, ESTABLISHING HEAT SINK. **2)** **IF** an ELAP is in progress **AND** CST is **NOT** available, **THEN** perform FSG-2, ALTERNATE AFW SUCTION SOURCE. 3) **IF** an ELAP is in progress and an alternate low pressure feedwater source is required due to 1P-29 unavailability, **THEN** perform the following: a) **IF** 2P-29 AND associated suction source is available, **THEN** perform FSG-15 UNIT 1, CROSS TIE AFW. b) **IF** 2P-29 AND associated suction source NOT available, **THEN** establish low pressure feedwater per ATTACHMENT L, ESTABLISHING LOW PRESSURE FEEDWATER FLOW.
	b. Control AFW Flow To Maintain S/G Levels Between - [51%] 32% and 63%	

****

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT M
(Page 3 of 7)
ASYMMETRICAL COOLDOWN

M4 Check DC Bus Loads:

a. Check vital instrumentation - AVAILABLE

- Any DC Bus voltage greater than 107.5 Vdc
- Vital instruments - REQUIRED INSTRUMENTS AVAILABLE:
  - o Red Instrument Bus Powered
  - o White Instrument Bus Powered

RESPONSE NOT OBTAINED: a. Perform FSG-7 UNIT 1, LOSS OF VITAL INSTRUMENTATION OR CONTROL POWER while continuing with this procedure.

****

M5 Maintain RCS Inventory:

a. Check PZR level - GREATER THAN [32%] 13%

RESPONSE NOT OBTAINED: a. Maintain level per FSG-1 Unit 1, LONG TERM RCS INVENTORY CONTROL.

****

M6 Check Boration Complete:

a. Check RCS borated to target value.

b. Time completed ____________

M7 Check Elapsed Time Since Boric Acid Addition Was Completed:

a. Check one hour has elapsed since M6.b

RESPONSE NOT OBTAINED: a. DO NOT CONTINUE until one hour has elapsed.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT M
(Page 4 of 7)
ASYMMETRICAL COOLDOWN

NOTES

- **Depressurization of S/G will result in SI actuation. SI should be reset to permit manual loading of equipment on AC safeguards bus.**
- **PZR level may be lost and reactor vessel upper head voiding may occur due to depressurization of S/G. Depressurization should not be stopped to prevent these occurrences.**

M8 **Depressurize Intact S/G to 320 psig:**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	a. Check S/G narrow range levels greater than [51%] 32% in available S/G	a. Perform the following: 1) Maintain maximum AFW flow until narrow range level greater than [51%] 32% in available S/G 2) **IF** an ELAP is in progress and an alternate low pressure feedwater source is required due to 1P-29 unavailability, **THEN** perform the following: a) **IF** 2P-29 AND associated suction source is available, **THEN** perform FSG-15 UNIT 1, CROSS TIE AFW. b) **IF** 2P-29 AND associated suction source NOT available, **THEN** establish low pressure feedwater per ATTACHMENT L, ESTABLISHING LOW PRESSURE FEEDWATER FLOW. 3) **WHEN** narrow range level is greater than [51%] 32% in at least one S/G, **THEN** continue with Step M8.b.

(Step M8, continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT M
(Page 5 of 7)
ASYMMETRICAL COOLDOWN

(Step M8. continued from previous page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	b. Manually dump steam using S/G ADVs to maintain cooldown rate in RCS cold legs less than 25°F/hr. o 1MS-2016 o 1MS-2015	b. Locally dump steam o 1MS-2016 o 1MS-2015
	c. Check S/G pressures - LESS THAN 320 psig	c. Perform the following: 1) **WHEN** S/G pressures decrease to less than 320 psig, **THEN** go to Step M8.d.
	d. Manually control S/G ADVs at 320 psig: o 1MS-2016 o 1MS-2015	d. Locally control S/G ADVs to maintain S/G pressure at 320 psig **IF** S/G pressure can **NOT** be maintained at 320 psig, **THEN** perform FSG-9 UNIT 1, LOW DECAY HEAT TEMPERATURE CONTROL to stop the uncontrolled cooldown
M9	**Check SI Signal Status:** • SI - HAS BEED ACTUATED	
M10	**Reset SI**	
M11	**Isolate SI Accumulators** a. Perform FSG-10 UNIT 1, PASSIVE RCS INJECTION ISOLATION. b. Verify Accumulators - ISOLATED • 1SI-841A • 1SI-841B	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT M
(Page 6 of 7)
ASYMMETRICAL COOLDOWN

NOTE

PZR level may be lost and reactor vessel upper head voiding may occur due to depressurization of S/G. Depressurization should not be stopped to prevent these occurrences.

M12 Depressurize Intact S/G to 150 psig:

a. Check S/G narrow range levels greater than [51%] 32% in available S/G

a. Perform the following:

1) Maintain maximum AFW flow until narrow range level greater than [51%] 32% in available S/G

2) **IF** an ELAP is in progress and an alternate low pressure feedwater source is required due to 1P-29 unavailability, **THEN** perform the following:

a) **IF** 2P-29 AND associated suction source is available, **THEN** perform FSG-15 UNIT 1, CROSS TIE AFW.

b) **IF** 2P-29 AND associated suction source NOT available, **THEN** establish low pressure feedwater per ATTACHMENT L, ESTABLISHING LOW PRESSURE FEEDWATER FLOW.

b. Manually dump steam using S/G ADVs to maintain cooldown rate in RCS cold legs less than 25°F/hr.

- 1MS-2016
- 1MS-2015

b. Locally dump steam

(Step M12 continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT M
(Page 7 of 7)
ASYMMETRICAL COOLDOWN

(Step M12. continued from previous page)

	c. Check S/G pressures - LESS THAN 150 psig	c. Perform the following: 1) **WHEN** S/G pressures decrease to less than 150 psig, **THEN** go to Step M12.d.
	d. Manually control S/G ADVs at 150 psig: o 1MS-2016 o 1MS-2015	d. Locally control S/G ADVs to maintain S/G pressure at 150 psig. **IF** S/G pressure can NOT be maintained at 150 psig, **THEN** perform FSG-9 UNIT 1, LOW DECAY HEAT TEMPERATURE CONTROL to stop the uncontrolled cooldown.
M13	Go To Step 70.	

-END-

FOLDOUT PAGE FOR ECA-0.0 UNIT 1

1 POWER RESTORATION CRITERIA

- o IF power is restored to any 480 Vac safeguards bus prior to placing ECCS components in pull out, **THEN** go to Step 46.
- o IF power is restored to any 480 Vac safeguards bus after ECCS components have been placed in pull out, **THEN** go to Step 70.

2 AFW SUPPLY SWITCHOVER CRITERIA

IF CST level lowers to less than 4 ft., **THEN** switch to alternate AFW suction supply per AOP-23 UNIT 1, ESTABLISHING ALTERNATE AFW SUCTION SUPPLY.

3 FAULTED S/G ISOLATION CRITERIA

IF any S/G pressure is trending lower in an uncontrolled manner OR any S/G completely depressurized, THEN the following may be performed:

a. Isolate feed flow to faulted S/G.

b. Maintain total feed flow greater than or equal to 230 gpm until narrow range level in at least one S/G is greater than [51%] 32%

4 ADVERSE CONTAINMENT CONDITIONS

IF any condition listed below occurs, **THEN** Environmentally Qualified (EQ) equipment and adverse containment setpoint values in brackets [ ], shall be used:

- o Containment pressure - GREATER THAN 5 psig
- o Containment radiation level - GREATER THAN OR EQUAL TO 1.0E+4 R/HR
- o Integrated dose to containment - GREATER THAN OR EQUAL TO 3.5E+4 R

APPLICABLE ONLY DURING ELAP CONDITIONS

5 ALTERNATE AFW SUCTION

CST is NOT Available OR CST level - LESS THAN 4 feet, THEN perform FSG-2, ALTERNATE AFW SUCTION SOURCE

6 ALTERNARE LOW PRESSURE FEEDWATER FLOW

Perform ATTACHMENT L if TDAFW flow is lost and is NOT immediately recoverable after Step 5 is performed.

7 LOSS OF VITAL INSTRUMENTATION OR CONTROL POWER

Perform FSG-7, LOSS OF VITAL INSTRUMENTATION OR CONTROL POWER if ELAP is in progress and EITHER condition listed below occurs:

- o DC Bus Voltages are less than 111 Vdc
- o Required vital instruments can NOT be energized

8 ALTERNATE CST MAKEUP

Perform FSG-6, ALTERNATE CST MAKEUP if CST level - LESS THAN 15.75 feet and ALL conditions listed below occur:

- CST is available
- AFW Flow has been verified

A. PURPOSE

1. This procedure provides directions to respond to a diesel malfunction, when an automatic start signal is present and the diesel does not respond as required.

2. This procedure is applicable for all plant conditions.

B. SYMPTOMS OR ENTRY CONDITIONS

1. Safeguard bus 1A-05 has experienced an undervoltage and the aligned diesel has failed to energize the bus as required.

C. REFERENCES

1. EC 283586, Transition to 10 CFR 50.48(c) - NFPA 805 From App R

2. EPM Report R2168-1003C-001

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

When closing electrical breakers remotely, the control switches should be held in the closed position for approximately 2 seconds to ensure UV relays have time to energize.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check Plant Stable Per AOP-18B Unit 1, TRAIN "B" EQUIPMENT OPERATION**	**IF NOT** previously performed, **THEN** go to AOP-18B Unit 1, TRAIN "B" EQUIPMENT OPERATION while continuing with this procedure as a secondary priority.
2	**Check Unit 1 - AT POWER**	**IF** Unit 1 shutdown **AND** both main feed pumps secured, **THEN** locally bypass the AMSAC actuation circuit using key #81: a. In Cable Spreading Room at panel 1N16, place bypass switch to - BYPASS b. Go to Step 6.

NOTE

When 1Y06 is deenergized, outward rod motion is blocked in both auto and manual modes.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Shift Control Rod Bank Selector Switch To - MANUAL**	
4	**Shut MSR Steam Supply Valves:** a. Adjust controller output to - ZERO • 1HC-2085	
5	**Shift IRPI To Alternate Power Supply:** a. Place Rod Position Indication Power Transfer switch in - ALT 1Y02	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	**Energize Bus 1A-05 From Diesel G-01:**	
	a. Check annunciator UNIT 1 4.16 kV BUS LOCKOUT - CLEAR • C02 D 3-4	a. Perform the following: 1) At cubicle 1A00-61, check 4.16 kV bus lockout on 1A-05. 2) **IF** 1A-05 is locked out, **THEN**: a) Consult with Maintenance to determine and correct cause. b) Return to Procedure And Step In Effect.
	b. Check G-01 to bus 1A-05 breaker - IN AUTO • 1A52-60	b. Go to Step 10.
	c. Check G-01 - RUNNING	c. Start G-01: 1) Ensure G-01 diesel mode selector switch - IN AUTO. 2) Turn G-01 diesel generator control switch to - START. 3) **IF** G-01 will **NOT** start, **THEN** go to Step 10.
	d. Ensure 1A-03 to 1A-05 bus tie breaker - OPEN • 1A52-57	
	e. Check G-01 to bus 1A-05 breaker - CLOSED • 1A52-60	e. Perform the following: 1) Try to auto-close breaker by placing control switch to trip position and then release. 2) **IF** breaker will **NOT** auto-close, **THEN**: a) Place G-01 mode selector switch to - EXERCISE. b) Turn synch switch for G-01 to bus 1A-05 breaker - ON. c) Manually close G-01 to bus 1A-05 breaker. • 1A52-60
7	**Check Bus 1A-05 - ENERGIZED**	Go to Step 10.
8	**Refer To TS 3.8, Electrical Power Systems**	
9	**Go To AOP-18A Unit 1, TRAIN "A" EQUIPMENT OPERATION**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

In fire scenarios, G-02 is susceptible to automatic tripping on overload when it is supplying both 1A-05 and 2A-05 due to spurious equipment operations.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	**Energize Bus 1A-05 From Diesel G-02:**	
	a. Check G-02 - RUNNING	a. Start G-02: 1) Ensure G-02 diesel mode selector switch - IN AUTO. 2) Turn G-02 diesel generator control switch to - START. 3) **IF** G-02 will **NOT** start, **THEN** go to Step 14.
	b. Ensure 1A-03 to 1A-05 bus tie breaker - OPEN • 1A52-57	
	c. Ensure G-01 to bus 1A-05 breaker - IN PULLOUT • 1A52-60	
	d. Using key #43, unlock and place G-02 to bus 1A-05 breaker control switch in AUTO • 1A52-66	
	e. Check G-02 to bus 1A-05 breaker - CLOSED • 1A52-66	e. Perform the following: 1) Try to auto-close breaker by placing control switch to trip position and then release. 2) **IF** breaker will **NOT** auto-close, **THEN**: a) Place G-02 mode selector switch to - EXERCISE. b) Turn synch switch for G-02 to bus 1A-05 breaker - ON c) Manually close G-02 to bus 1A-05 breaker. • 1A52-66

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	**Check Bus 1A-05 - ENERGIZED**	Go to Step 14.
12	**Refer To TS 3.8, Electrical Power Systems**	
13	**Go To AOP-18A Unit 1, TRAIN "A" EQUIPMENT OPERATION**	
14	**Check Bus H-02 - ENERGIZED**	**IF** bus H-01 energized **OR** bus H-03 energized, **THEN** perform the following: a. **IF** annunciator 13.8 kV BUS **OR** FEEDER LOCKOUT in alarm, **THEN** go to Step 16. • C02 E 2-9 b. Ensure bus H-02 normal feed - OPEN • H52-20 c. Ensure H-03 to H-01 bus tie breaker - CLOSED • H52-31 d. Turn synch switch for H-02 to H-01 bus tie breaker - ON e. Close H-02 to H-01 bus tie breaker. • H52-21 **IF** bus H-02 can **NOT** be energized, **THEN** go to Step 16.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15	**Restore Power To 1A-03:**	
	a. Check annunciator UNIT 1 4.16 kV BUS LOCKOUT - CLEAR • C02 D 3-4	a. Perform the following: 1) At cubicle 1A52-40, check 4.16 kV bus lockout on 1A-03. 2) **IF** 1A-03 is locked out, **THEN** select emergency diesel power supply: o ATTACHMENT A, G-01 LOCAL MANUAL START o ATTACHMENT B, G-02 LOCAL MANUAL START
	b. Check annunciator 1X-04 LOW VOLTAGE STATION AUX TRANS LOCKOUT - CLEAR • C02 D 1-7	b. Go to Step 16.
	c. Reset and close bus H-02 feed to 1X-04 breaker • H52-22	c. Go to Step 16.
	d. Turn synch switch for bus 1A-03 normal feed breaker - ON	
	e. Reset and close bus 1A-03 normal feed breaker • 1A52-36	e. Go to Step 16.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16	Check Bus 1A-03 - ENERGIZED	Perform the following: a. **IF** bus 1A-01 is powered from 1X-02, **THEN** energize bus 1A-03 from 1A-01: 1) Ensure bus 1A-03 normal feed - OPEN • 1A52-36 2) Ensure 1A-03 to 2A-03 bus tie breaker - OPEN • 1A52-40 3) Turn synch switch for 1A-03 to 1A-01 bus tie breaker - ON 4) Close 1A-03 to 1A-01 bus tie breaker. • 1A52-37 5) Turn synch switch for 1A-03 to 1A-01 bus tie breaker - OFF 6) **IF** power is restored to 1A-03, **THEN** go to step 17. b. **IF** bus 2A-03 is powered from 2X-04, **THEN** energize bus 1A-03 from 2A-03: 1) Ensure bus 1A-03 normal feed - OPEN • 1A52-36 2) Ensure 1A-03 to 1A-01 bus tie breaker - OPEN • 1A52-37 3) Turn synch switch for 1A-03 to 2A-03 bus tie breaker - ON 4) Close 1A-03 to 2A-03 bus tie breaker. • 1A52-40 5) Turn synch switch for 1A-03 to 2A-03 bus tie breaker - OFF c. **IF** 1A-03 can **NOT** be powered from 1A-01 or 2A-03, **THEN** select emergency diesel power supply: o ATTACHMENT A, G-01 LOCAL MANUAL START o ATTACHMENT B, G-02 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17	**Energize 1A-05 From 1A-03:**	
	a. Ensure G-01 to bus 1A-05 breaker - OPEN • 1A52-60	
	b. Ensure G-02 to bus 1A-05 breaker - OPEN • 1A52-66	
	c. Turn synch switch for 1A-03 to 1A-05 bus tie breaker - ON	
	d. Reset and close 1A-03 to 1A-05 bus tie breaker • 1A52-57	d. Select emergency diesel power supply: o ATTACHMENT A, G-01 LOCAL MANUAL START <u>OR</u> o ATTACHMENT B, G-02 LOCAL MANUAL START
18	**Check Bus 1A-05 - ENERGIZED**	Return to <u>Step 6</u>.
19	**Refer To TS 3.8, Electrical Power Systems**	
20	**<u>Go To AOP-18A Unit 1, TRAIN "A" EQUIPMENT OPERATION</u>**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 1 of 5)
G-01 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A1	**Check Green "Power On" Light - ENERGIZED** • Panel C-64A • Panel C-34	**IF** green light **NOT** lit, **THEN** transfer control power to alternate source: a. At PAB 8' elevation South of Unit 2 charging pumps, direct PAB operator to shut switch D31-01. b. At C-78, shift to alternate power by swapping paired breakers: • For annunciators, open breaker 1 and close breaker 2. • For start circuit 1, open breaker 3 and close breaker 4. • For control power, open breaker 5 and close breaker 6. • For field flash, open breaker 7 and close breaker 8.
A2	**Check Overspeed Trip Alarms - CLEAR** • Panel C-64A • Panel C-34	Reset mechanical overspeed trip and alarms: a. Place mode selector switch in - LOCAL b. Reset mechanical overspeed trip. c. Place mode selector switch in - AUTO

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 2 of 5)
G-01 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A3	**Emergency Start G-01:**	
	a. At C-34A, place local/remote transfer switches to - LOCAL • Transfer switch No. 1 • Transfer switch No. 2	
	b. At C-34A, start G-01 by depressing EMERGENCY START push-button	b. **IF** G-01 will **NOT** emergency start, **THEN** manually start G-01: 1) At C-64, place mode selector switch in - LOCAL START. 2) At C-64, depress and hold ENGINE START push-button until engine speed rises to idle. 3) At C-64, raise engine speed to 900 rpm by depressing idle release push-button. 4) **IF** G-01 can **NOT** be started, **THEN** do not continue and inform Control Room of G-01 status.
A4	**At C-64, Check G-01 Speed - GREATER THAN OR EQUAL TO 900 RPM**	Perform the following: a. At C-64, place governor mode switch to - "HYD" b. **IF** diesel speed **NOT** greater than or equal to 900 rpm, **THEN** at C-64, raise speed using hydraulic governor control switch.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 3 of 5)
G-01 LOCAL MANUAL START

CAUTION

Rubber gloves with leather protectors and personnel safety equipment are required to locally operate FFC relay.

A5 Check G-01 Frequency - BETWEEN 59.7 Hz HZ AND 60.3 Hz

Perform the following:

a. **IF** field **NOT** flashed, **THEN** in C-34, manually actuate relay FFC for 3 seconds by pushing yellow tab on relay actuator located behind terminal connections.

b. At C-64, ensure 43/G-01-GOV, G01 EDG Mode Selector Switch in HYD, **THEN** adjust frequency using the hydraulic governor control switch.

c. **IF** hydraulic governor control switch **NOT** functional, **THEN** adjust frequency using SPEED control knob on faceplate of Woodward governor.

d. **IF** frequency can **NOT** be maintained, **THEN** locally shutdown G-01:

1) At C-64, push both engine stop pushbuttons.

2) Pull fuel supply cut-off valve operator.

3) At C-34A, place output breaker in pullout:

- 1A52-60

4) Go to ATTACHMENT B, G-02 LOCAL MANUAL START.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 4 of 5)
G-01 LOCAL MANUAL START

A6	**Contact Control Room To Check G-01 Voltage - BETWEEN 4050 VAC AND 4300 VAC**	**IF** field is **NOT** flashed, **THEN** in C-34, manually actuate relay FFC for 3 seconds by pushing yellow tab on relay actuator located behind terminal connections. Contact Control Room to maintain voltage between 4050 Vac and 4300 Vac by adjusting diesel loading. a. **IF** voltage can **NOT** be maintained, **THEN** locally shutdown G-01: b. Push both engine stop push-buttons. c. Pull fuel supply cut-off valve operator. d. At C-34A, place output breaker in pullout. • 1A52-60 e. Go to ATTACHMENT B, G-02 LOCAL MANUAL START.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 5 of 5)
G-01 LOCAL MANUAL START

A7	**Energize Bus 1A-05 From Normal Diesel Supply G-01:**	
	a. Check G-01 - RUNNING	a. Go to ATTACHMENT B, G-02 LOCAL MANUAL START.
	b. Locally ensure 1A-03 to 1A-05 bus tie breaker - OPEN • 1A52-57	b. Go to ATTACHMENT B, G-02 LOCAL MANUAL START.
	c. Locally ensure G-01 to 2A-05 bus tie breaker control switch - IN PULLOUT • 1A52-73	c. Go to ATTACHMENT B, G-02 LOCAL MANUAL START.
	d. Locally ensure G-01 to bus 1A-05 breaker control switch - IN AUTO • 1A52-60	d. **Place** G-01 to bus 1A-05 breaker control switch - IN AUTO
	e. Locally check G-01 to bus 1A-05 breaker - CLOSED • 1A52-60	e. **Locally**: 1) Try to auto-close breaker by placing control switch to trip position then release. 2) **IF** breaker will **NOT** auto-close, **THEN** manually close breaker control switch.
A8	**Go To AOP-18A Unit 1, TRAIN "A" EQUIPMENT OPERATION**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 1 of 5)
G-02 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B1	**Dispatch Operator With Key #43 To G-02**	
B2	**Check Green "Power On" Light - ENERGIZED** • Panel C-65A • Panel C-35	IF green light NOT lit, THEN transfer control power to alternate source: a. At PAB 8' elevation South of Unit 2 charging pumps, direct PAB operator to shut switch D41-01. b. At C-79, shift to alternate power by swapping paired breakers: • For annunciators, open breaker 1 and close breaker 2. • For start circuit 1, open breaker 3 and close breaker 4. • For control power, open breaker 5 and close breaker 6. • For field flash, open breaker 7 and close breaker 8.
B3	**Check Overspeed Trip Alarms - CLEAR** • Panel C-65A • Panel C-35	Reset mechanical overspeed trip and alarms: a. Place mode selector switch in - LOCAL b. Reset mechanical overspeed trip. c. Place mode selector switch - IN AUTO

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 2 of 5)
G-02 LOCAL MANUAL START

B4	**Emergency Start G-02:**	
	a. At C-35A, place local/remote transfer switches to - LOCAL • Transfer switch No. 1 • Transfer switch No. 2	
	b. At C-35A, start G-02 by depressing EMERGENCY START push-button	b. **IF** G-02 will **NOT** emergency start, **THEN** manually start G-02: 1) At C-65, place mode selector switch in - LOCAL START. 2) At C-65, depress and hold ENGINE START push-button until engine speed rises to idle. 3) At C-65, raise engine speed to 900 rpm by depressing idle release push-button. 4) **IF** G-02 can **NOT** be started, **THEN** do not continue and inform Control Room of G-02 status.
B5	**At C-65, Check G-02 Speed - GREATER THAN OR EQUAL TO 900 RPM**	Perform the following: a. At C-65, place governor mode switch to - "HYD" b. **IF** diesel speed **NOT** greater than or equal to 900 rpm, **THEN** at C-65, raise speed using hydraulic governor control switch.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 3 of 5)
G-02 LOCAL MANUAL START

CAUTION

Rubber gloves with leather protectors and personnel safety equipment are required to locally operate FFC relay.

B6	**Have the control room check G-02 Frequency - BETWEEN 59.7 Hz AND 60.3 Hz**	Perform the following: a. **IF** field **NOT** flashed, **THEN** in C-35, manually actuate relay FFC for 3 seconds by pushing yellow tab on relay actuator located behind terminal connections. b. At C-65, ensure 43/G-02-GOV, G02 EDG Mode Selector Switch in HYD, **THEN** adjust frequency using the hydraulic governor control switch. c. **IF** hydraulic governor control switch **NOT** functional, **THEN** adjust frequency using SPEED control knob on faceplate of Woodward governor. d. **IF** frequency can **NOT** be maintained, **THEN** locally shutdown G-02: 1) Push both engine stop pushbuttons. 2) Pull fuel supply cut-off valve operator. 3) At C-35A, place output breaker in pullout: • 1A52-66 4) Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 4 of 5)
G-02 LOCAL MANUAL START

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B7	**Check G-02 Voltage - BETWEEN 4050 VAC AND 4300 VAC**	**IF** field is **NOT** flashed, **THEN** in C-35, manually actuate relay FFC for 3 seconds by pushing yellow tab on relay actuator located behind terminal connections. Contact Control Room to maintain voltage between 4050 Vac and 4300 Vac by adjusting diesel loading. **IF** voltage can **NOT** be maintained, **THEN** locally shutdown G-02: a. Push both engine stop push-buttons. b. Pull fuel supply cut-off valve operator. c. At C-35A, place output breaker in pullout. • 1A52-66 d. Return to Procedure And Step In Effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 5 of 5)
G-02 LOCAL MANUAL START

B8	**Energize Bus 1A-05 From Alternate Diesel Supply G-02:**	
	a. Check G-02 - RUNNING	a. Return to Procedure And Step In Effect.
	b. Locally ensure 1A-03 to 1A-05 bus tie breaker - OPEN • 1A52-57	b. Return to Procedure And Step In Effect.
	c. Locally ensure G-02 to 2A-05 bus tie breaker control switch- IN PULLOUT • 2A52-67	c. Return to Procedure And Step In Effect.
	d. Locally unlock and place G-02 to bus 1A-05 breaker control switch - IN AUTO • 1A52-66	
	e. Locally check G-02 to bus 1A-05 breaker - CLOSED • 1A52-66	e. Locally: 1) Try to auto-close breaker by placing control switch to trip position then release. 2) **IF** breaker will **NOT** auto-close, **THEN** manually close breaker control switch.
B9	**Go To AOP-18A Unit 1, TRAIN "A" EQUIPMENT OPERATION**	

-END-

ATTACHMENT 5

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 286, ADOPTION OF EMERGENCY ACTION LEVEL SCHEME PURSUANT TO NEI 99-01 REVISION 6,
"DEVELOPMENT OF EMERGENCY ACTION LEVELS FOR NON-PASSIVE REACTORS"

UPDATED PBNP EAL SCHEME WALLBOARDS

3 pages follow

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

R — Abnormal Rad Levels / Rad Effluent

1 — Rad Effluent

General Emergency: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE [pg. 36]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RG1.1
Reading on **ANY** Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer.

RG1.2
Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY

RG1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 1000 mR/hr expected to continue for greater than or equal to 60 min.
- Analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for 60 min. of inhalation.

Site Area Emergency: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE [pg. 33]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RS1.1
Reading on **ANY** Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer.

RS1.2
Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY

RS1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 100 mR/hr expected to continue for greater than or equal to 60 min.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for 60 min. of inhalation.

Alert: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE [pg. 27]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RA1.1
Reading on **ANY** Table R-1 effluent radiation monitor greater than column "ALERT" for 15 minutes or longer.

RA1.2
Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond SITE BOUNDARY.

RA1.3
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.

RA1.4
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

Unusual Event: Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer [pg. 23]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RU1.1
Reading on **ANY** Table R-1 effluent radiation monitor greater than column "UE" for 60 minutes or longer.

RU1.2
Reading on ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.3
Sample analyses for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODCM limits for 60 minutes or longer.

2 — Irradiated Fuel Event

General Emergency: Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer [pg. 38]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RG2.1
Spent fuel pool level cannot be restored to at least 40 ft. 8 in. for 60 minutes or longer.

Site Area Emergency: Spent fuel pool level at the top of the fuel racks [pg. 30]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RS2.1
Lowering of spent fuel pool level to 40 ft. 8 in.

Alert: Significant lowering of water level above, or damage to, irradiated fuel [pg. 29]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by a reading on **ANY** of the following radiation monitors greater than the value shown:
- RE-105 SFP Area Low Range Radiation Monitor 4 R/hr
- 1(2)-RE-126 Containment High Radiation Monitor 7 R/hr
- 1(2)-RE-127 Containment High Radiation Monitor 7 R/hr
- 1(2)-RE-128 Containment High Radiation Monitor 7 R/hr

RA2.3
Lowering of spent fuel pool level to 49 ft.0 in.

Unusual Event: Unplanned loss of water level above irradiated fuel [pg. 25]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RU2.1
a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:
- Spent fuel pool low water level alarm
- Visual observation

AND

b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors.
- RE-105 SFP Area Low Range Radiation Monitor
- RE-135 SFP Area High Range Radiation Monitor
- 1(2) RE-102 El. 66' CONTAINMENT Low Range Monitor

Table R-1 Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	UE
Gaseous	1(2)-RE-307 CTMNT Purge Exhaust Mid Range Gas with only containment purge in operation	-----	6.0E+1 µCi/cc	6.0E+0 µCi/cc	1.4E-2 µCi/cc
Gaseous	1(2)-RE-309 CTMNT Purge Exhaust High Range Gas with only containment purge in operation	6.0E+2 µCi/cc	6.0E+1 µCi/cc	6.0E+0 µCi/cc	-----
Gaseous	2-RE-305 CTMNT Purge Exhaust Low Range Gas with both purge and GS building ventilation in operation	-----	-----	-----	9.4E-3 µCi/cc
Gaseous	2-RE-307 CTMNT Purge Exhaust Mid Range Gas with both purge and GS building ventilation in operation	-----	4.0E+1 µCi/cc	4.0E+0 µCi/cc	9.4E-3 µCi/cc
Gaseous	2-RE-307 CTMNT Purge Exhaust Mid Range Gas with only GS building ventilation in operation	-----	-----	-----	2.8E-2 µCi/cc
Gaseous	2-RE-307 CTMNT Purge Exhaust Mid Range Gas with only forced vent of containment	-----	-----	-----	1.0E+1 µCi/cc
Gaseous	2-RE-309 CTMNT Purge Exhaust High Range Gas with both purge and GS building ventilation in operation	4.0E+2 µCi/cc	4.0E+1 µCi/cc	4.0E+0 µCi/cc	-----
Gaseous	2-RE-309 CTMNT Purge Exhaust High Range Gas with only GS building ventilation in operation	1.2E+3 µCi/cc	1.2E+2 µCi/cc	1.2E+1 µCi/cc	-----
Gaseous	2-RE-309 CTMNT Purge Exhaust High Range Gas with only forced vent of containment	-----	-----	-----	1.0E+1 µCi/cc
Gaseous	RE-315 AB Exhaust Low Range Gas	-----	-----	-----	5.4E-3 µCi/cc
Gaseous	RE-317 AB Exhaust Mid Range Gas	1.0E+2 µCi/cc	1.0E+1 µCi/cc	1.0E+0 µCi/cc	5.4E-3 µCi/cc
Gaseous	RE-319 AB Exhaust High Range Gas	1.0E+2 µCi/cc	1.0E+1 µCi/cc	1.0E+0 µCi/cc	-----
Gaseous	RE-325 Drumming Area Exhaust Low Range Gas	-----	-----	-----	8.4E-3 µCi/cc
Gaseous	RE-327 Drumming Area Exhaust Mid Range Gas	-----	1.6E+1 µCi/cc	1.6E+0 µCi/cc	8.4E-3 µCi/cc
Liquid	1(2)-RE-229 Service Water Overboard	-----	-----	-----	2.3E-3 µCi/cc

3 — Area Radiation Levels

Alert: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown [pg. 31]

1 | 2 | 3 | 4 | 5 | 6 | DEF

RA3.1
Dose rate greater than 15 mR/hr in **ANY** of the following areas:
- Control Room (RE-101)
- Central Alarm Station **AND** Secondary Alarm Station (by survey)

RA3.2
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to any Table R-2 plant rooms or areas.

Table R-2 Safe Ops, S/D, C/D Areas

Room/Area	Mode
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 / 4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B32 MCC Area	4

E — ISFSI

1 — Confinement Boundary

Table E-1 Cask On-Contact Dose Rates

32 PT DSC		VSC-24	
HSM Front	1700 mrem/hr	Sides	200 mrem/hr
HSM Door	400 mrem/hr	Top	400 mrem/hr
End Shield Wall Ext.	12 mrem/hr	Air Inlets	700 mrem/hr
		Air Outlets	200 mrem/hr

Unusual Event: Damage to a loaded cask CONFINEMENT BOUNDARY [pg. 61]

1 | 2 | 3 | 4 | 5 | 6 | DEF

EU1.1
Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than the values shown on Table E-1 on the surface of the spent fuel cask.

H — Hazards

1 — Security

General Emergency: HOSTILE ACTION resulting in loss of physical control of the facility [pg. 99]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HG1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor
AND
EITHER of the following has occurred:
- ANY of the following safety functions cannot be controlled or maintained
  - Reactivity control
  - Core cooling
  - RCS heat removal

  OR
- Damage to spent fuel has occurred or is IMMINENT

Site Area Emergency: HOSTILE ACTION within the PROTECTED AREA [pg. 96]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HS1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor

Alert: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes [pg. 99]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Unusual Event: Confirmed SECURITY CONDITION or threat [pg. 79]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HU1.1
A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the Security Shift Supervisor.

HU1.2
Notification of a credible security threat directed at PBNP.

HU1.3
A validated notification from the NRC providing information of an aircraft threat.

2 — Seismic Event

Unusual Event: Seismic event greater than OBE level [pg. 81]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HU2.1
Seismic event greater than Operating Basis Earthquake (OBE) as indicated by seismic monitor indication of ground acceleration greater than 0.06 g horizontal **OR** 0.04 g vertical.

3 — Natural or Tech. Hazard

Unusual Event: Hazardous event [pg. 82]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HU3.1
A tornado strike within the PROTECTED AREA

HU3.2
Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

HU3.3
Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

HU3.4
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

HU3.5
Lake level greater than or equal to +8.0 ft. (Plant elevation)

HU3.6
Pump bay level less than -15.0 ft.

4 — Fire

Table H-1 Fire Areas
- Control Room
- Containment
- PAB
- G05 building
- 13.8kV Building
- Cable Spreading Room
- Vital Switchgear Room
- AFW Pump Room
- G-01/02 Rooms
- EDG Building
- Service Water Pump Rooms
- Facade 85'

Unusual Event: FIRE potentially degrading the level of safety of the plant [pg. 84]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HU4.1
A FIRE is **not** extinguished within 15 min. of **ANY** of the following FIRE detection indications:
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within **ANY** Table H-1 plant rooms or areas

HU4.2
Receipt of a single fire alarm with no other indications of a FIRE
AND
The FIRE is located within ANY Table H-1 plant rooms or areas except Containment in Modes 1 and 2
AND
The existence of a FIRE is not verified within 30 minutes of alarm receipt.

HU4.3
A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60 minutes of the initial report, alarm or indication.

HU4.4
A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

5 — Hazardous Gases

Alert: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown [pg. 91]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HA5.1
Release of a toxic, corrosive, asphyxiant or flammable gas into any Table H-2 plant rooms or areas
AND
Entry into the room or area is prohibited or impeded.

Table H-2 Safe Ops, S/D, C/D Areas

Room/Area	Mode
U1 VCT Area	3 / 4 / 5
U2 VCT Area	3 / 4 / 5
U1 Primary Sample area	3
U2 Primary Sample area	3
CCW HX Room	4 / 5
C-59 area	3 / 4 / 5
Pipeway 2, 8 ft. Elev.	3 / 4
Pipeway 3, 8 ft. Elev.	3 / 4
1/2B32 MCC Area	4

6 — Control Room Evacuation

Site Area Emergency: Inability to control a key safety function from outside the Control Room [pg. 97]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HS6.1
An event has resulted in plant control being transferred from the Control Room to AOP local control stations.
AND
Control of **ANY** of the following key safety functions is not reestablished within 15 minutes
- Reactivity
- Core cooling
- RCS heat removal

Alert: Control Room evacuation resulting in transfer of plant control to alternate locations [pg. 93]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HA6.1
An event has resulted in plant control being transferred from the Control Room to AOP local control stations.

7 — ED Judgment

General Emergency: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of General Emergency [pg. 101]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HG7.1
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Site Area Emergency: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of Site Area Emergency [pg. 98]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HS7.1
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Alert: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of an Alert [pg. 94]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HA7.1
Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Unusual Event: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE [pg. 88]

1 | 2 | 3 | 4 | 5 | 6 | DEF

HU7.1
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Modes:

1	2	3	4	5	6	DEF
Power Operation	Startup	Hot Standby	Hot Shutdown	Cold Shutdown	Refueling	Defueled

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
S System Malfunct. — **1** Loss of Emergency AC Power	Prolonged loss of all offsite and all onsite AC power to safeguard buses [pg. 126] 1 | 2 | 3 | 4 **SG1.1** Loss of ALL offsite and ALL onsite AC power to 1(2)-A05 and 1(2)-A06. **AND** **EITHER:** • Restoration of at least one AC emergency bus in less than 4 hours is not likely. • Conditions requiring entry into Core Cooling – Red Path (CSP-C.1) are met.	Loss of all offsite and all onsite AC power to safeguard buses for 15 minutes or longer [pg. 123] 1 | 2 | 3 | 4 **SS1.1** Loss of **ALL** offsite and **ALL** onsite AC power to 1(2)-A05 and 1(2)-A06 for 15 minutes or longer.	Loss of all but one AC power source to safeguard buses for 15 minutes or longer [pg. 114] 1 | 2 | 3 | 4 **SA1.1** AC power capability to 1(2)-A05 **AND** 1(2)-A06 is reduced to a single power source for 15 minutes or longer. **AND** Any additional single power source failure will result in a loss of ALL AC power to SAFETY SYSTEMS.	Loss of all offsite AC power capability to safeguard buses for 15 minutes or longer [pg. 103] 1 | 2 | 3 | 4 **SU1.1** Loss of **ALL** offsite AC power capability to 1(2)-A05 and 1(2)-A06 for 15 minutes or longer.
2 Loss of Vital DC Power	Loss of all AC and Vital DC power sources for 15 minutes or longer [pg. 129] 1 | 2 | 3 | 4 **SG2.1** Loss of **ALL** offsite and **ALL** onsite AC power capability power to 1(2)-A05 and 1(2)-A06 for 15 minutes or longer. **AND** Indicated voltage is less than 115 VDC on **ALL** Vital DC busses D-01, D-02, D-03 and D-04 for 15 minutes or longer.	Loss of all Vital DC power for 15 minutes or longer [pg. 124] 1 | 2 | 3 | 4 **SS2.1** Indicated voltage is less than 115 VDC on **ALL** Vital DC busses D-01, D-02, D-03, and D-04 for 15 minutes or longer.		
3 Loss of Control Room Indications			UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress [pg. 116] 1 | 2 | 3 | 4 **SA3.1** An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for 15 minutes or longer **AND** **Any** of the Table S-2 transient events are in progress	UNPLANNED loss of Control Room indications for 15 minutes or longer [pg. 104] 1 | 2 | 3 | 4 **SU3.1** An UNPLANNED event results in the inability to monitor one or more of the Table S-1 parameters from within the Control Room for 15 minutes or longer.
4 RCS Activity		**Table S-2 Significant Transients** • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor trip • SI actuation	**Table S-1 Safety System Parameters** • Reactor power • RCS / Pressurizer Level • RCS / Pressurizer Pressure • Core Exit / RCS Temperature • Level in at least one steam generator • Steam Generator Auxiliary Feed Water Flow	Reactor coolant activity greater than Technical Specification allowable limits [pg. 106] 1 | 2 | 3 | 4 **SU4.1** Failed Fuel Monitor 1(2)-RE-109 reading greater than 750 mR/hr. **SU4.2** Sample analysis indicates that a RCS Specific Activity value is greater than an allowable limit specified in Technical Specifications as indicated by **ANY** of the following conditions: a. Dose Equivalent I-131 greater than 50 µCi/gm **OR** b. Dose Equivalent I-131 greater than 0.5 µCi/gm but less than or equal to 50 µCi/gm for greater than 48 hours **OR** c. Dose Equivalent Xe-133 greater than 300 µCi/gm for greater than 48 hours
5 RCS Leakage				RCS leakage for 15 minutes or longer [pg. 107] 1 | 2 | 3 | 4 **SU5.1** RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer. **OR** RCS identified leakage greater than 25 gpm for 15 minutes or longer. **OR** Leakage from the RCS to a location outside containment, or Steam Generator tube leakage, greater than 25 gpm for 15 minutes or longer.
6 RPS Failure		Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal [pg. 125] 1 | 2 | 3 | 4 **SS6.1** An automatic or manual trip did not shut down the reactor **AND** All actions to shut down the reactor have been unsuccessful **AND** **EITHER**: • Conditions requiring entry into Core Cooling – Red Path (CSP-C.1) are met. • Conditions requiring entry into Heat Sink – Red Path (CSP-H.1) are met.	Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor [pg. 118] 1 | 2 | 3 | 4 **SA6.1** An automatic or manual trip fails to shut down the reactor as indicated by reactor power greater than 5% **AND** Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	Automatic or manual trip fails to shut down the reactor [pg. 109] 1 | 2 | 3 | 4 **SU6.1** An automatic trip did not shut down the reactor **AND** A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. **SU6.2** A manual trip did not shutdown the reactor. **AND** **EITHER** of the following: • A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor **OR** • A subsequent automatic trip is successful in shutting down the reactor.
7 Loss of Comm.				Loss of all onsite or offsite communications capabilities [pg. 111] 1 | 2 | 3 | 4 **SU7.1** Loss of **ALL** of the following onsite communication methods: • Plant Public Address System (Gai-Tronics) • Commercial Phones • PBX Phones • Security Radio • Portable Radios **SU7.2** Loss of **ALL** of the following offsite response organization communications methods: • Nuclear Accident Reporting System (NARS) • Commercial Phones • PBX Phones • Satellite Phones • Manitowoc County Sheriff's Department Radio **SU7.3** Loss of **ALL** of the following NRC communications methods: • FTS Phone System • Commercial Phones • PBX Phones • Satellite Phones
8 CMT Failure				Failure to isolate containment or loss of containment pressure control [pg. 112] 1 | 2 | 3 | 4 **SU8.1** Failure of containment to isolate when required by an actuation signal. **AND** **ALL** required penetrations are not closed within 15 minutes of the actuation signal. **SU8.2** Containment pressure greater than 25 psig. **AND** Less than one full train of Containment Cooling System equipment is operating per design for 15 minutes or longer.
9 Hazardous Event Affecting Safety Systems		**Table S-3 Hazardous Events** • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Lake level greater than or equal to +9.0 ft. (Plant elevation) • Pump bay level less than -19.0 ft. • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode [pg. 120] **SA9.1** The occurrence of **ANY** of the Table S-3 hazardous events: **AND** • Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode. **AND** • **EITHER** of the following: • Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or • The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.	
F Fission Product Barrier Degradation	**FG1** 1 | 2 | 3 | 4 Loss of any two barriers and Loss or Potential Loss of third barrier (Table F-1) [pg. 63]	**FS1** 1 | 2 | 3 | 4 Loss or potential loss of any two barriers (Table F-1) [pg. 63]	**FA1** 1 | 2 | 3 | 4 Any Loss or any Potential Loss of either Fuel Clad or RCS barrier (Table F-1) [pg. 63]	None

Table F-1 Fission Product Barrier Matrix

Category	Fuel Clad (FC) Barrier [pg. 66]		Reactor Coolant System (RCS) Barrier [pg. 68]		Containment (CNTMT) Barrier [pg. 71]	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A Critical Safety Function Status	Conditions requiring entry into Core Cooling **RED** Path (CSP C.1) are met.	Conditions requiring entry into Core Cooling **ORANGE** Path (CSP C.2) are met. **OR** Conditions requiring entry into Heat Sink **RED** Path (CSP H.1) are met.	None	Conditions requiring entry into Heat Sink **RED** Path (CSP H.1) are met. **OR** Conditions requiring entry into RCS Integrity **RED** Path (CSP P.1) are met.	None	Conditions requiring entry into Core Cooling **RED** Path (CSP C.1) are met. **AND** CSP C.1 not effective within 15 minutes.
B RCS or SG Tube Leakage	None	None	An automatic or manual ECCS (SI) actuation required by **EITHER** of the following: • UNISOLABLE RCS leakage **OR** • SG tube RUPTURE	Operation of a standby charging (makeup) pump is required by **EITHER** of the following: • UNISOLABLE RCS leakage **OR** • SG tube leakage	A leaking or RUPTURED SG is FAULTED outside of containment.	None
C RCS Activity / Containment Radiation	Containment radiation monitor reading greater than 577 R/hr indicated on **ANY** of the following: • 1(2)-RE-126 • 1(2)-RE-127 • 1(2)-RE-128 **OR** 1(2)-RE-109 greater than 4,500 mR/hr	None	Containment radiation monitor reading greater than 11 R/hr indicated on **ANY** of the following: • 1(2)-RE-126 • 1(2)-RE-127 • 1(2)-RE-128	None	None	Containment radiation monitor reading greater than 18,500 R/hr indicated on **ANY** of the following: • 1(2)-RE-126 • 1(2)-RE-127 • 1(2)-RE-128
D Containment Integrity or Bypass	None	None	None	None	Containment isolation is required **AND EITHER** of the following: • Containment integrity has been lost based on Emergency Coordinator judgment **OR** • UNISOLABLE pathway from containment to the environment exists. **OR** Indications of RCS leakage outside of containment	Containment pressure greater than 60 psig **OR** 6% H^2 inside containment. **OR** • Containment pressure greater than 25 psig. **AND** • Less than one full train of depressurization equipment is operating per design for 15 minutes or longer.
E ED Judgment	**ANY** condition in the opinion of the Emergency Coordinator that indicates Loss of the Fuel Clad barrier.	**ANY** condition in the opinion of the Emergency Coordinator that indicates Potential Loss of the Fuel Clad barrier.	**ANY** condition in the opinion of the Emergency Coordinator that indicates Loss of the RCS barrier.	**ANY** condition in the opinion of the Emergency Coordinator that indicates Potential Loss of the RCS barrier.	**ANY** condition in the opinion of the Emergency Coordinator that indicates Loss of the Containment barrier.	**ANY** condition in the opinion of the Emergency Coordinator that indicates Potential Loss of the Containment barrier.

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refueling | DEF Defueled

NEXTera ENERGY RESOURCES

Point Beach Nuclear Plant
EAL Classification Matrix
Page 2 of 3
HOT CONDITIONS
(RCS > 200°F)

C Cold SD/ Refueling System Malfunct.

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

1 RCS Level

GENERAL EMERGENCY

Loss of RCS inventory affecting fuel clad integrity with Containment challenged [pg. 57]

Modes: 5 6

CG1.1

Reactor vessel/RCS level cannot be monitored for 30 minutes or longer.

AND

Core uncovery is indicated by ANY of the following:

- Containment High Radiation Monitors (1(2)-RE-126, RE-127, or RE-128) reading greater than 100 R/hr
- Erratic source range monitor indication
- UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery

AND

ANY indication from the Containment Challenge Table C-1.

Table C-1 Containment Challenge

- CONTAINMENT CLOSURE not established *
- 6% H2 exists inside containment
- UNPLANNED increase in containment pressure

* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

SITE AREA EMERGENCY

Loss of RCS inventory affecting core decay heat removal capability [pg. 55]

Modes: 5 6

CS1.1

Reactor vessel/RCS level cannot be monitored for 30 minutes or longer.

AND

Core uncovery is indicated by **ANY** of the following:

- Containment High Radiation Monitor (1(2)-RE-126, RE-127, or RE-128) reading greater than 100 R/hr
- Erratic source range monitor indication
- UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery

ALERT

Loss of RCS inventory [pg. 48]

Modes: 5 6

CA1.1

Loss of reactor vessel/RCS inventory as indicated by level less than 16% on LI-447 / LI-447A.

CA1.2

Reactor vessel/RCS level cannot be monitored for 15 minutes or longer

AND

UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels due to a loss of reactor vessel/RCS inventory.

UNUSUAL EVENT

UNPLANNED loss of RCS inventory for 15 minutes or longer [pg. 40]

Modes: 5 6

CU1.1

UNPLANNED loss of reactor coolant results in reactor vessel/ RCS level less than a required lower limit for 15 minutes or longer.

CU1.2

RCS water level cannot be monitored

AND

UNPLANNED increase in Containment Sump A **OR** Waste Holdup Tank levels

2 Loss of Emergency AC Power

ALERT

Loss of all offsite and all onsite AC power to safeguard buses for greater than 15 minutes [pg. 50]

Modes: 5 6 DEF

CA2.1

Loss of **ALL** offsite and **ALL** onsite AC Power to 1(2)-A05 and 1(2)-A06 for 15 minutes or longer.

UNUSUAL EVENT

Loss of all but one AC power source to safeguard buses for 15 minutes or longer [pg. 42]

Modes: 5 6 DEF

CU2.1

AC power capability to 1(2)-A05 and 1(2)-A06 is reduced to a single power source for 15 minutes or longer.

AND

Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.

3 RCS Temp.

Table C-2 RCS Heat-up Duration Thresholds

RCS Status	Containment Closure Status	Heat-up Duration
Intact (but not REDUCED INVENTORY)	N/A	60 min.*
Not intact OR REDUCED INVENTORY	Established	20 min.*
	Not established	0 min.

* If an RHR is in operation within this time frame and RCS temperature is being reduced the EAL is not applicable.

ALERT

Inability to maintain plant in cold shutdown [pg. 51]

Modes: 5 6

CA3.1

UNPLANNED increase in RCS temperature to greater than 200°F for greater than the duration specified in Table C-2.

CA3.1

UNPLANNED RCS pressure increase greater than 25 psig. (This EAL does not apply during water-solid plant conditions.)

UNUSUAL EVENT

UNPLANNED increase in RCS temperature [pg. 44]

Modes: 5 6

CU3.1

UNPLANNED increase in RCS temperature to greater than 200°F

CU3.2

Loss of **ALL** RCS temperature and reactor vessel/RCS level indication for 15 minutes or longer.

4 Loss of Vital DC Power

UNUSUAL EVENT

Loss of vital DC power for 15 minutes or longer [pg. 46]

Modes: 5 6

CU4.1

Indicated voltage is less than 115 VDC on required Vital DC buses D-01, D-02, D-03, D-04 for 15 minutes or longer.

5 Loss of Comm.

UNUSUAL EVENT

Loss of all onsite or offsite communications capabilities [pg. 47]

Modes: 5 6 DEF

CU5.1

Loss of **ALL** of the following onsite communication methods:

- Plant Public Address System (Gai-Tronics)
- Commercial Phones
- PBX Phones
- Security Radio
- Portable Radios

CU5.2

Loss of **ALL** of the following offsite response organization communications methods:

- Nuclear Accident Reporting System (NARS)
- Commercial Phones
- PBX Phones
- Satellite Phones
- Manitowoc County Sheriff's Department Radio

CU5.3

Loss of **ALL** of the following NRC communications methods:

- FTS Phone System
- Commercial Phones
- PBX Phones
- Satellite Phones

6 Hazardous Event Affecting Safety Systems

Table C-3 Hazardous Events

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Lake level greater than or equal to +9.0 ft. (Plant elevation)
- Pump bay level less than -19.0 ft.
- Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

ALERT

Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode [pg. 53]

Modes: 5 6

CA6.1

The occurrence of **ANY** of the Table C-3 hazardous events.

AND

- Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode.

  AND
- **EITHER** of the following:
  - Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
  - The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refueling | DEF Defueled

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