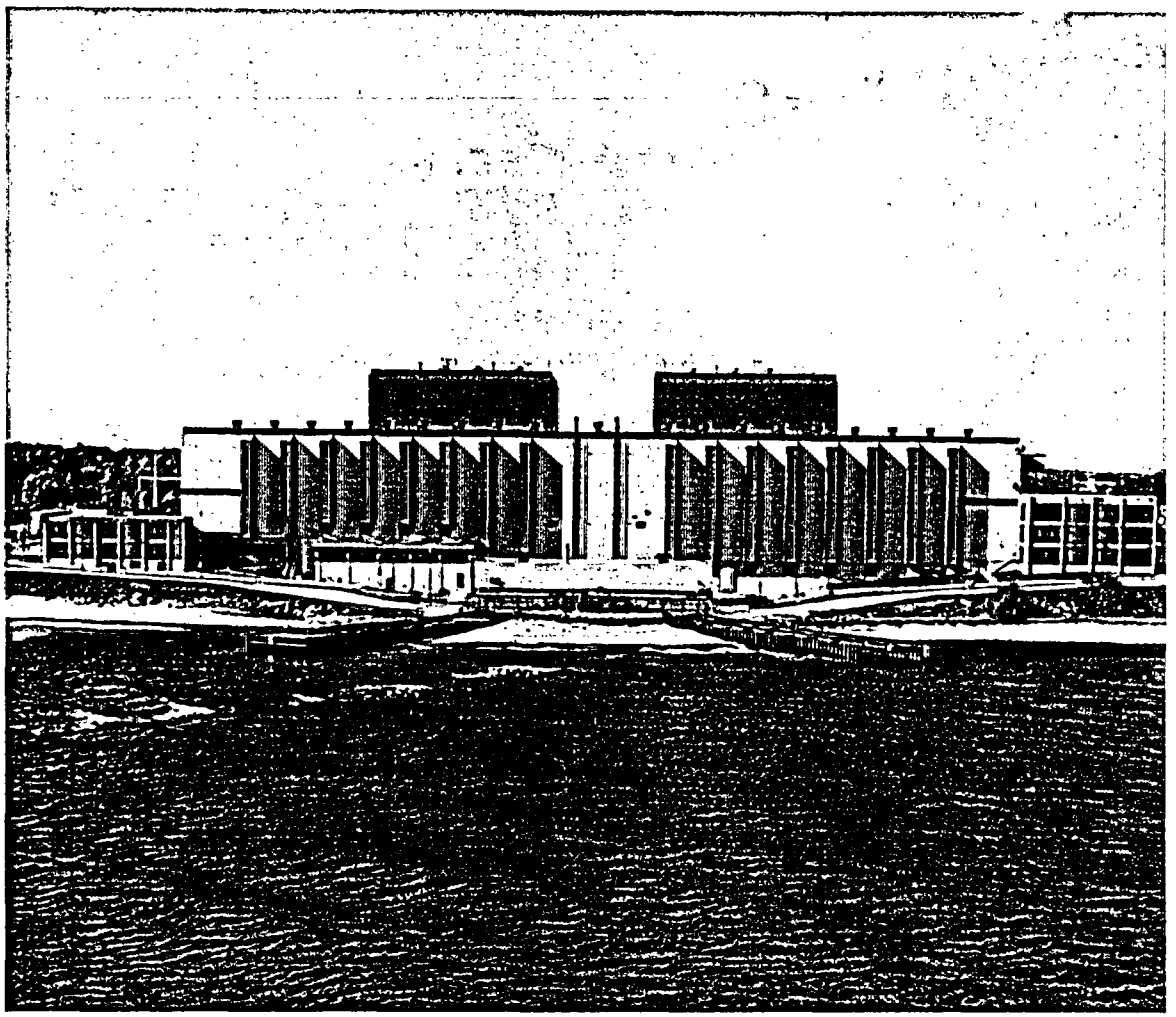
TREAT AS
SENSITIVE
INFORMATION

ENCLOSURE IV
ATTACHMENTS 1 & 2



POINT BEACH
REVISED EAL SUBMITTAL

ATTACHMENT 1: RED LINE OF THE TECHNICAL BASIS DOCUMENT

ATTACHMENT 2: CLEAN COPY OF THE TECHNICAL BASIS DOCUMENT

Enclosure IV

Attachment 1: Red-Line Technical Basis Document

Point Beach Nuclear Plant

Emergency Action Level Technical Basis Document

Emergency Action Level Technical Bases Document

Reviewed by;

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F - Fission Product Barrier Degradation F-1

H - Hazards H-1

S - System Malfunction S-1

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ACRONYMS

AC	Alternating Current
ATWS	Anticipated Transient Without Scram
CCW	Component Cooling Water
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
CMT	Containment
CSF	Critical Safety Function
CSFST	Critical Safety Function Status Tree
DC	Direct Current
DHR	Decay Heat Removal
DOT	Department of Transportation
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
EPIP	Emergency Plan Implementing Procedure
EPRI	Electric Power Research Institute
ERG	Emergency Response Guideline
ESF	Engineered Safeguards Feature
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
GE	General Emergency
HPSI	High Pressure Safety Injection
IC	Initiating Condition
IDLH	Immediately Dangerous to Life and Health
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI	Independent Spent Fuel Storage Installation
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LFL	Lower Flammability Limit

LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection
MSIV	Main Steam Isolation Valve
mR	milliRoentgen
mrem	milliRem
Mw	Megawatt
NEI	Nuclear Energy Institute
NESP	National Environmental Studies Project
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
PSIG	Pounds per Square Inch Gauge
R	Roentgen
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specifications
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RVLIS	Reactor Vessel Level Indicating System
SAE	Site Area Emergency
SG	Steam Generator
SI	Safety Injection
SPDS	Safety Parameter Display System
SRO	Senior Reactor Operator
SSE	Safe Shutdown Earthquake
TEDE	Total Effective Dose Equivalent
TOAF	Top of Active Fuel
TSC	Technical Support Center
UE	Notification Of Unusual Event
WE	Westinghouse Electric
WOG	Westinghouse Owners Group

1. PURPOSE

This document provides the detailed set of Emergency Action Levels (EALs) applicable to the Point Beach Nuclear Plant (PBNP) and the associated Technical Bases using the EAL development methodology found in NEI 99-01 Revision 4 [Ref. 2.1]. Personnel responsible for implementation of EPIP-1.2 (Emergency Classification) [Ref. 2.2], and the Emergency Action Level Matrix [Ref. 2.3] may use this document as a technical reference and an aid in EAL interpretation.

2. REFERENCES

2.1 NEI 99-01 Revision 4, Methodology for Development of Emergency Action Levels

2.2 EPIP-1.2 (Emergency Classification)

2.3 Emergency Action Level Matrix

2.4 PBNP Technical Specifications Table 1.1-1

3. DISCUSSION

3.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the PBNP Emergency Plan.

In 1992, the NRC endorsed NUMARC/NESP-007 "Methodology for Development of Emergency Action Levels" as an alternative to NUREG 0654 EAL guidance.

NEI 99-01 (NUMARC/NESP-007) Revision 4 represents the most recently NRC endorsed methodology per RG 1.101 Rev 4, "Emergency Planning and Preparedness for Nuclear Power Reactors." Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.
- Addressing initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations.
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Using NEI 99-01 Rev. 4, PBNP conducted an EAL implementation upgrade project that produced the EALs discussed herein. While the upgraded EALs are site-specific, an objective of the project was to ensure to the extent possible EAL conformity and consistency between the NMC plant sites.

3.2 Key Definitions in EAL Methodology

The following definitions apply to the generic EAL methodology:

EMERGENCY CLASS: One of a minimum set of names or titles, established by the Nuclear Regulatory Commission (NRC), for grouping of normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time sensitive onsite and off site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called:

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

Section 3.3 provides further discussion of the emergency classes.

INITIATING CONDITION (IC): One of a predetermined subset of nuclear power plant conditions when either the potential exists for a radiological emergency, or such an emergency has occurred.

- An IC is an emergency condition which sets it apart from the broad class of conditions that may or may not have the potential to escalate into a radiological emergency.
- It can be a continuous, measurable function that is outside technical specifications, such as elevated RCS temperature or falling reactor coolant level (a symptom).
- It also encompasses occurrences such as FIRE (an event) or reactor coolant pipe failure (an event or a barrier breach).

EMERGENCY ACTION LEVEL (EAL): A pre determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

- There are times when an EAL will be a threshold point on a measurable continuous function, such as a primary system coolant leak that has exceeded technical specifications.
- At other times, the EAL and the IC will coincide, both identified by a discrete event that places the plant in a particular emergency class.

3.3 Recognition Categories

ICs and EALs are grouped in one of several categories. This classification scheme incorporates symptom-based, event-based, and barrier-based ICs and EALs.

- R - Abnormal Rad Levels/Radiological Effluent
- C - Cold Shutdown./ Refueling System Malfunction
- E - Independent Spent Fuel Storage Installation (ISFSI)
- F - Fission Product Barrier Degradation
- H - Hazards
- S - System Malfunction

Some recognition categories are further divided into one or more subcategories depending on the types and number of plant conditions that dictate emergency classifications. An EAL may or may not exist for each subcategory at all four classification levels. Similarly, more than one EAL may exist for a subcategory in a given emergency classification when appropriate (i.e., no EAL at the General Emergency level but three EALs at the Unusual Event level).

ICs and EALs can be grouped in one of several schemes. This generic classification scheme incorporates symptom-based, event-based, and barrier-based ICs and EALs.

The symptom-based category for ICs and EALs refers to those indicators that are measurable over some continuous spectrum, such as core temperature, coolant levels, containment pressure, etc. When one or more of these indicators begin to show off-normal readings, reactor operators are trained to identify the probable causes and potential consequences of these "symptoms" and take corrective action. The level of seriousness indicated by these symptoms depends on the degree to which they have exceeded technical specifications, the other symptoms or events that are occurring contemporaneously, and the capability of the licensed operators to gain control and bring the indicator back to safe levels.

Event-based EALs and ICs refer to occurrences with potential safety significance, such as the failure of a high-pressure safety injection pump, a safety valve failure, or a loss of electric power to some part of the plant. The range of seriousness of these "events" is dependent on the location, number of contemporaneous events, remaining plant safety margin, etc.

Barrier-based EALs and ICs refer to the level of challenge to principal barriers used to assure containment of radioactive materials contained within a nuclear power plant. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers concurrently under challenge. In reality, barrier-based EALs are a subset of symptom-based EALs that deal with symptoms indicating fission product barrier

challenges. These barrier-based EALs are primarily derived from Emergency Operating Procedure (EOP) Critical Safety Function (CSF) Status Tree Monitoring (or their equivalent). Challenge to one or more barriers generally is initially identified through instrument readings and periodic sampling. Under present barrier-based EALs, deterioration of the reactor coolant system pressure boundary or the fuel clad barrier usually indicates an "Alert" condition, two barriers under challenge a Site Area Emergency, and loss of two barriers with the third barrier under challenge is a General Emergency. The fission product barrier matrix described in Section 5-F is a hybrid approach that recognizes that some events may represent a challenge to more than one barrier, and that the containment barrier is weighted less than the reactor coolant system pressure boundary and the fuel clad barriers.

Symptom-based ICs and EALs are most easily identified when the plant is in a normal startup, operating or hot shutdown mode of operation, with all of the barriers in place and the plant's instrumentation and emergency safeguards features fully operational as required by technical specifications. It is under these circumstances that the operations staff has the most direct information of the plant's systems, displayed in the main control room. As the plant moves through the decay heat removal process toward cold shutdown and refueling, barriers to fission products are reduced (i.e., reactor coolant system pressure boundary may be open) and fewer of the safety systems required for power operation are required to be fully operational. Under these plant operating modes, the identification of an IC in the plant's operating and safety systems becomes more event-based, as the instrumentation to detect symptoms of a developing problem may not be fully effective; and engineered safeguards systems, such as the Emergency Core Cooling System (ECCS), are partially disabled as permitted by the plant's Technical Specifications.

Barrier-based ICs and EALs also are heavily dependent on the ability to monitor instruments that indicate the condition of plant operating and safety systems. Fuel cladding integrity and reactor coolant levels can be monitored through several indicators when the plant is in a normal operating mode, but this capability is much more limited when the plant is in a refueling mode, when many of these indicators are disconnected or off-scale. The need for this instrumentation is lessened, however, and alternate instrumentation is placed in service when the plant is shut down.

It is important to note that in some operating modes there may not be definitive and unambiguous indicators of containment integrity available to control room personnel. For this reason, barrier-based EALs should not place undue reliance on assessments of containment integrity in all operating modes. Generally, Technical Specifications relax maintaining containment integrity requirements in modes 5 and 6 in order to provide flexibility in performance of specific tasks during shutdown conditions. Containment pressure and temperature indications may not increase if there is a pre-existing breach of containment integrity. At most plants, a large portion of the containment's exterior cannot be monitored for leakage by radiation monitors.

Several categories of emergencies have no instrumentation to indicate a developing problem, or the event may be identified before any other indications are recognized. A reactor coolant pipe could break; FIRE alarms could sound; radioactive materials could be released; and any number of other events can occur that would place the plant in an

emergency condition with little warning. For emergencies related to the reactor system and safety systems, the ICs shift to an event based scheme as the plant mode moves toward cold shutdown and refueling modes. For non-radiological events, such as FIRE, external floods, wind loads, etc., as described in NUREG-0654 Appendix 1, event-based ICs are the norm.

In many cases, a combination of symptom-, event- and barrier-based ICs will be present as an emergency develops. In a loss of coolant accident (LOCA), for example:

- Coolant level is dropping; (symptom)
- There is a leak of some magnitude in the system (pipe break, safety valve stuck open) that exceeds plant capabilities to make up the loss; (barrier breach or event)
- Core (coolant) temperature is rising; (symptom) and
- At some level, fuel failure begins with indicators such as high off-gas, high coolant activity samples, etc. (barrier breach or symptom)

To the extent possible, the EALs are symptom based. That is, the action level is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. But, a purely symptom based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

Category R - Abnormal Rad Levels/Radiological Effluent and Category F - Fission Product Barrier Degradation are primarily symptom-based. The symptoms are indicative of actual or potential degradation of either fission product barriers or personnel safety.

Other categories tend to be event-based. For example, System Malfunctions are abnormal and emergency events associated with vital plant system failures, while Hazards are those non-plant system related events that have affected or may affect plant safety.

3.4 Emergency Class Descriptions

There are three considerations related to the emergency classes. These are:

- The potential impact on radiological safety, either as now known or as can be reasonably projected.
- How far the plant is beyond its predefined design, safety and operating envelopes.
- Whether or not conditions that threaten health are expected to be confined to within the site boundary.

The ICs deal explicitly with radiological safety affect by escalating from levels corresponding to releases within regulatory limits to releases beyond EPA Protective Action Guideline (PAG) plume exposure levels.

UNUSUAL EVENT: Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

- Potential degradation of the level of safety of the plant is indicated primarily by exceeding plant technical specification Limiting Condition of Operation (LCO) allowable action statement time for achieving required mode change.

- Precursors of more serious events may be included because precursors represent a potential degradation in the level of safety of the plant.

- Minor releases of radioactive materials are included. In this emergency class, however, releases do not require monitoring or offsite response (e.g., dose consequences of less than 10 millirem).

ALERT: Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

- Rather than discussing the distinguishing features of "potential degradation" and "potential substantial degradation," a comparative approach would be to determine whether increased monitoring of plant functions is warranted at the Alert level as a result of safety system degradation. This addresses the operations staff's need for help, independent of whether an actual decrease in plant safety is determined. This increased monitoring can then be used to better determine the actual plant safety state, whether escalation to a higher emergency class is warranted, or whether de-escalation or termination of the emergency class declaration is warranted. Dose consequences from these events are small fractions of the EPA PAG plume exposure levels, i.e., about 10 millirem to 100 millirem TEDE.

SITE AREA EMERGENCY: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for 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.

- The discriminator (threshold) between Site Area Emergency and General Emergency is whether or not the EPA PAG plume exposure levels are expected to be exceeded outside the site boundary.

- This threshold, in addition to dynamic dose assessment considerations discussed in the EAL guidelines, clearly addresses NRC and offsite emergency response agency concerns as to timely declaration of a General Emergency.

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. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

- The bottom line for the General Emergency is whether evacuation or sheltering of the general public is indicated based on EPA PAGs and, therefore, should be interpreted to include radionuclide release regardless of cause.

- To better assure timely notification, EALs in this category are primarily expressed in terms of plant function status, with secondary reliance on dose projection. In terms of fission product barriers, loss of two barriers with loss or potential loss of the third barrier constitutes a General Emergency.

3.5 Emergency Class Thresholds

The most common bases for establishing these boundaries are the technical specifications and setpoints for each plant that have been developed in the design basis calculations and the Final Safety Analysis Report (FSAR).

For those conditions that are easily measurable and instrumented, the boundary is likely to be the EAL (observable by plant staff, instrument reading, alarm setpoint, etc.) that indicates entry into a particular emergency class. For example, the main steam line radiation monitor may detect high radiation that triggers an alarm. That radiation level also may be the setpoint that closes the main steam isolation valves (MSIV) and initiates the reactor scram. This same radiation level threshold, depending on plant-specific parameters, also may be the appropriate EAL for a direct entry into an emergency class.

In addition to the continuously measurable indicators, such as coolant temperature, coolant levels, leak rates, containment pressure, etc., the FSAR provides indications of the consequences associated with design basis events. Examples would include steam pipe breaks, MSIV malfunctions, and other anticipated events that, upon occurrence, place the plant immediately into an emergency class.

Another approach for defining these boundaries is the use of a plant-specific probabilistic safety assessment (PSA - also known as probabilistic risk assessment, PRA). PSAs have been completed for several individual plants, but this is by no means comprehensive. There are, however, PSAs that have been completed for representative plant types such as is done in NUREG-1150, "Severe Accident Risks: An Assessment for Five Nuclear Power Plants," as well as several other utility-sponsored PSAs. Existing PSAs can be used as a good first approximation of the relevant ICs and risk associated with emergency conditions for existing plants.

Another critical element of the analysis to arrive at these threshold (boundary) conditions is the time that the plant might stay in that condition before moving to a higher emergency class. In particular, station blackout coping analyses performed in response to 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout," may be used to determine whether a specific plant enters a Site Area Emergency or a General Emergency directly, and when escalation to General Emergency is indicated. The time dimension is critical to the EAL since the purpose of the emergency class for state and local officials is to notify them of

the level of mobilization that may be necessary to handle the emergency. This is particularly true when a "Site Area Emergency" or "General Emergency" is imminent. Establishing EALs for such conditions must take estimated evacuation time into consideration to minimize the potential for the plume to pass while evacuation is underway.

Regardless of whether or not containment integrity is challenged, it is possible for significant radioactive inventory within containment to result in EPA PAG plume exposure levels being exceeded even assuming containment is within technical specification allowable leakage rates. With or without containment challenge, however, a major release of radioactivity requiring offsite protection actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

3.6 Emergency Action Levels

With the emergency classes defined, the thresholds that must be met for each EAL to be placed under the emergency class can be determined. There are two basic approaches to determining these EALs. EALs and emergency class boundaries coincide for those continuously measurable, instrumented ICs, such as radioactivity, core temperature, coolant levels, etc. For these ICs, the EAL will be the threshold reading that most closely corresponds to the emergency class description using the best available information.

For discrete (discontinuous) events, the approach will have to be somewhat different. Typically, in this category are internal and external hazards such as FIRE or earthquake. The purpose for including hazards in EALs is to assure that station personnel and offsite emergency response organizations are prepared to deal with consequential damage these hazards may cause. If, indeed, hazards have caused damage to safety functions or fission product barriers, this should be confirmed by symptoms or by observation of such failures. Therefore, it may be appropriate to enter an Alert status for events approaching or exceeding design basis limits such as Operating Basis Earthquake, design basis wind loads, FIRE within VITAL AREAs, etc. This would give the operating staff additional support and improved ability to determine the extent of plant damage. if damage to barriers or challenges to Critical Safety Functions (CSFs) have occurred or are identified, then the additional support can be used to escalate or terminate the Emergency Class based on what has been found. Of course, security events must reflect potential for increasing security threat levels.

Plant emergency operating procedures (EOPs) are designed to maintain and/or restore a set of CSFs which are listed in the order of priority for restoration efforts during accident conditions. While the actual nomenclature of the CSFs may vary among plants, generally the PWR CSF set includes:

- Subcriticality
- Core cooling
- Heat sink
- Pressure-temperature-stress (RCS integrity)

- Containment
- RCS inventory

There are diverse and redundant plant systems to support each CSF. By monitoring the CSFs instead of the individual system component status, the impact of multiple events is inherently addressed, e.g., the number of operable components available to maintain the critical safety function.

The EOPs contain detailed instructions regarding the monitoring of these functions and provides a scheme for classifying the significance of the challenge to the functions. In providing EALs based on these schemes, the emergency classification can flow from the EOP assessment rather than being based on a separate EAL assessment. This is desirable as it reduces ambiguity and reduces the time necessary to classify the event.

As an example, consider that the Westinghouse Owner's Group (WOG) Emergency Response Guidelines (ERGs) classify challenges as YELLOW, ORANGE, and RED paths. If the core exit thermocouples exceed 1200 degrees F or 700 degrees F with low reactor vessel water level, a RED path condition exists. The ERG considers a RED path as "... an extreme challenge to a plant function necessary for the protection of the public ..." This is almost identical to the present NRC NUREG-0654 description of a site area emergency "... actual or likely failures of plant functions needed for the protection of the public ..." It reasonably follows that if any CSF enters a RED path, a site area emergency exists. A general emergency could be considered to exist if core cooling CSF is in a RED path and the EOP function restoration procedures have not been successful in restoring core cooling.

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds has been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

3.7 Treatment of Multiple Events And Emergency Class Upgrading

The above discussion deals primarily with simpler emergencies and events that may not escalate rapidly. However, usable EAL guidance must also consider rapidly evolving and complex events. Hence, emergency class upgrading and consideration of multiple events must be addressed.

There are three approaches presently in use for covering multiple events and emergency class upgrading. These approaches are:

(U1) Multiple contemporaneous events are counted and are the basis for escalating to a higher emergency class. For example, two or more contemporaneous Alerts escalate to a Site Area Emergency.

(U2) The emergency class is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency.

(U3) Emergency Director judgment. Although all emergency classifications require judgment, some utilities rely on Emergency Director judgment with little or no additional explicit guidance.

An additional approach for plants with PRAs is to make use of event tree analysis to define combinations of events which lead to equivalent risks. Such event sequences should have an equal emergency classification assigned. However, the chief drawback to this approach as well as (U1) above, is that multiple events may be masked when they actually occur. Further, for plants using symptom-based (and barrier-based) emergency procedures, direct perception of multiple events is unnecessary.

Emergency class upgrading for multi-unit stations with shared safety-related systems and functions must also consider the effects of a loss of a common system on more than one unit (e.g. potential for radioactive release from more than one core at the same site). For example, many two-unit stations have their control panels for both units in close proximity within the same room. Thus, control room evacuation most likely would affect both units. There are a number of other systems and functions which may be shared at a given multi-unit station. This must be considered in the emergency class declaration and in the development of appropriate site-specific ICs and EALs based on the generic EAL guidance.

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds has been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

PBNP will utilize the following methodology:

With appropriate consideration for Emergency Director judgment EALs, properly structured EALs on a fission product barrier basis and which include equivalent risk, will appropriately escalate multiple events to a higher emergency class. For example, common cause failures such as loss of ultimate heat sink or loss of all AC power, will result in multiple contemporaneous symptoms indicating safety system functional failures and increasing challenge to fission product barriers. It is the existence of these symptoms (barrier challenges) that escalate the emergency class, whether there are one or multiple causes

The emergency class is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency.

Since PBNP is a dual-unit plant, emergency class upgrading must consider the effects of a loss of a common system on the other unit. For example, the control panels for both units share the same room. Thus, control room evacuation most likely would affect both units. There are a number of other systems and functions which may be shared. This must be considered in the emergency class declaration.

3.8 Emergency Class Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing. There are several approaches presently in use for emergency class downgrading. These approaches are:

(D1) Terminate the emergency class declaration.

(D2) Recovery from emergency class.

(D3) Combination of downgrading approaches. Many utilities reviewed include the option to downgrade to a lower emergency class. This is consistent with actions called for in NUREG-0654 Appendix 1. However, these utilities state that their experience more closely resembles (D1) and (D2) above as practical choices.

Another approach possible with risk-based EALs is a relatively simple approach for upgrading to a higher emergency class when the risk increases and downgrading when risk decreases. The boundaries for emergency categories are defined in terms of risk in this approach, and discrete events fall into these categories based on risk. This means that within each emergency class, there is uniformity to the relative levels of risk to human health and safety from radiological accidents. However, this option may not be practical when applied to actual emergencies, especially those involving General Emergencies.

PBNP will utilize the following methodology:

> **A combination approach involving recovery from General Emergencies and Site Area Emergencies and termination from UEs, Alerts, causing no long-term plant damage appears to be the best choice. Downgrading to lower emergency classes adds notifications but may have merit under certain circumstances.**

3.9 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. For example, an emergency classification is warranted when automatic and manual actions taken within the control room do not result in a required reactor trip. However, it is likely that actions taken outside of the control room will be successful, probably before the Emergency Director classifies the event. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses (e.g., coolant sampling, may be necessary).

In general, observe the following guidance: Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event

and other termination criteria are met. For example, a momentary event, such as an ATWS or an earthquake, requires declaration even though the condition may have been resolved by the time the declaration is made.

- An ATWS represents a failure of a front line safety system (RPS) designed to protect the health and safety of the public.
- The affect of an earthquake on plant equipment and structures may not be readily apparent until investigations are conducted.

There may be cases in which a plant condition that exceeded an EAL threshold was not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Rev. 1, Section 3 should be applied.

3.10 Operating Mode Applicability

Technical Specifications [Ref. 2.4] provides definitions for the following operating modes:

1 Power Operations

K_{eff} is GREATER THAN OR EQUAL TO 0.99 and reactor power is GREATER THAN 5% rated thermal power.

2 Startup

K_{eff} is GREATER THAN OR EQUAL TO 0.99 and reactor power is LESS THAN OR EQUAL TO 5% rated thermal power.

3 Hot Standby

K_{eff} is LESS THAN 0.99 and average reactor coolant temperature (Tavg) GREATER THAN OR EQUAL TO 350°F.

4 Hot Shutdown

K_{eff} is LESS THAN 0.99 and average reactor coolant temperature (Tavg) LESS THAN 350°F and GREATER THAN 200°F.

5 Cold Shutdown

K_{eff} is LESS THAN 0.99 and average reactor coolant temperature (Tavg) LESS THAN OR EQUAL TO 200°F.

6 Refuel

One or more vessel head closure bolts less than fully tensioned

In addition to the Technical Specification operating modes, NEI 99-01 [Ref. 1] defines the following additional mode:

D Defueled

All reactor fuel removed from Reactor Vessel (full core off load during refueling or extended outage)

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action is initiated in response to the condition) should be compared to the mode applicability of the EALs. If a lower or higher plant operating mode is reached before the emergency classification is made, the declaration shall be based on the mode that existed at the time the event occurred.

Recognition categories are associated with the operating modes listed in the following matrix:

	Recognition Category					
Mode	R	C	E	F	H	S
1 - Power Operations	X			X	X	X
2 - Startup	X			X	X	X
3 - Hot Standby	X			X	X	X
4 - Hot Shutdown	X			X	X	X
5 - Cold Shutdown	X	X			X	
6 - Refuel	X	X			X	
D - Defueled	X	X			X	
N/A			X			

3.11 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon loss of or potential loss to one or more of the three fission product barriers. "Loss" and "potential loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials and "potential loss" means imminent loss of the barrier.

The primary fission product barriers are:

- Fuel Cladding (FC): Zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the FC barrier.

- Reactor Coolant System (RCS): The reactor vessel shell, vessel head, vessel nozzles and penetrations and all primary systems directly connected to the reactor vessel up to the first containment isolation valve comprise the RCS barrier.

- Containment (CMT): The vapor containment structure and all isolation valves required to maintain containment integrity under accident conditions comprise the Containment barrier.

3.12 Emergency Classification Based on Fission Product Barrier Degradation

The following criteria are the bases for event classification related to fission product barrier loss or challenge:

- Unusual Event:

  Any loss or any potential loss of Containment

- Alert:

  Any loss or any potential loss of either Fuel Cladding or RCS

- Site Area Emergency:

  Loss or potential loss of any two barriers

- General Emergency:

  Loss of any two barriers and loss or potential loss of third barrier

3.13 EAL Relationship to EOPs and Critical Safety Function Status

Where possible, the EALs have been made consistent with and utilize the conditions defined in the PBNP Critical Safety Function Status Trees (CSFSTs). While the symptoms that drive operator actions specified in the CSFSTs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. Where these symptoms are clearly representative of one of the NEI Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the CSFSTs, classification of emergencies using these EALs is not dependent upon Emergency Operating Procedures (EOP) entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

3.14 Imminent EAL Thresholds

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds has been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes. Explicit EALs, specifying use of Emergency Director judgment, are given in the Hazards, ISFSI and Fission Product Barrier Degradation categories.

4. TECHNICAL BASES INFORMATION

4.1 Recognition Category Organization

The technical bases of the EALs are provided under Recognition Categories R, C, E, F, H and S of this document. A table summarizing the Initiating Conditions introduces each category. The tables provide an overview of how the ICs are related under each emergency class. ICs within each category are listed according to classification (as applicable) in the following order: Unusual Event, Alert, Site Area Emergency, and General Emergency.

For Recognition Category F, Table F-0 defines the emergency classifications associated with barrier loss and potential loss. Table F-1 lists the thresholds associated with the loss and potential loss of each fission product barrier. The presentation method shown for Table F-1 was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. Basis discussion of the thresholds immediately follows Table F-1.

4.2 Initiating Condition Structure

ICs in Recognition Categories R, C, E, H and S are structured in the following manner:

- Recognition Category Title
- IC Identifier:
    - First character identifies the category by letter (R, C, E, H and S)
    - Second character identifies the emergency classification level (UE for Unusual Event, A for Alert, S for Site Area Emergency, and G for General Emergency)
    - Third character is the numerical sequence as given in Revision 4 of NEI 99-01 [Ref. 1] (e.g., SA2). Due to document revisions, certain NEI ICs have been deleted, leaving gaps in the numerical sequence.

- Emergency Class: Unusual Event, Alert, Site Area Emergency, or General Emergency

- IC Description

- Operating Mode Applicability: Refers to the operating mode during which the IC/EAL is applicable

- Emergency Action Level(s): EALs are the conditions applicable to the criteria of the IC and are used to determine the need to classify an event/condition. If more than one EAL is applicable to an IC, emergency classification is required when any EAL within the IC reaches the EAL threshold. To clarify this intent, ICs with multiple EALs include a parenthetical phrase in the EAL title line, indicating that each constitutes an emergency classification. For example, the phrase "(RA1.1 or RA1.2)" indicates that either EAL is An Alert.

- Basis: Provides information that explains the IC and EAL(s). Plant source document references are provided as needed to substantiate site-specific information included in the EALs and bases.

4.3 EAL Identification

The EAL identifier is the IC identifier followed by a period and sequence number (e.g., RU1.1, RU1.2, etc.). If only one EAL is assigned to an IC, the EAL is given the number one.

The primary purpose of the EAL identifier is to uniquely distinguish each classifiable condition. Secondary purposes are to assist location of an EAL within the EAL classification scheme and to announce the emergency classification level.

5. EMERGENCY ACTION LEVEL BASES

The information is presented by Recognition Categories:

R - Abnormal Rad Levels/Radiological Effluent

- Radiological Effluents
- Abnormal Radiation Levels
- Irradiated Fuel Accidents

C - Cold Shutdown / Refueling System Malfunction

- Loss of AC Power
- Loss of DC Power
- Decay Heat Removal
- RCS Leakage/RCS Draindown
- Loss of Reactor Vessel Inventory
- Fuel Clad Degradation

- Loss of Communications
- Inadvertent Criticality

E - Independent Spent Fuel Storage Installation (ISFSI)

- Dry Fuel Storage

F - Fission Product Barrier Degradation

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

H – Hazards

- Security Events
- Control Room Evacuation
- Natural or man-Made Events
- FIRE/EXPLOSION
- Toxic or Flammable Gases
- Discretionary

S - System Malfunction

- Loss of AC Power
- Loss of DC Power
- Failure of Reactor Protection System (RPS)
- Decay Heat Removal
- Loss of Annunciators
- RCS Leakage
- Fuel Clad Degradation
- Loss of Communications
- Technical Specifications
- Inadvertent Criticality

The Initiating Conditions for each of the above Recognition Categories R, C, E, F, H, and S are in the order of UE, Alert, Site Area Emergency, and General Emergency. For all Recognition Categories, an Initiating Condition matrix versus Emergency Class is first shown. For Recognition Category F, the barrier-based EALs are presented in Table F-1. For all other Recognition Categories Initiating Condition matrices are not required. The

purpose of the IC matrices is to provide the reader with an overview of how the ICs are logically related under each Emergency Class.

Each of the EAL guides in Recognition Categories R, C, E, F, H, and S is structured in the following way:

- **Recognition Category** - As described above.

- **Emergency Class** - UE, Alert, Site Area Emergency or General Emergency.

- **Initiating Condition** – Symptom- or Event-Based, Generic Identification and Title.

- **Operating Mode Applicability** - refers to the operating mode (PWRs) during which the IC/EAL is applicable - Power Operation (includes Startup Mode in PWRs), Hot Standby (includes Hot Standby / Startup Condition in BWRs), Hot Shutdown, Cold Shutdown, Refueling, Defueled, All, or None. These modes are defined in each licensee's technical specifications. The mode classifications and terminology appropriate to the specific facility should be used. Note that Permanently Defueled and ISFSI IC/EALs have no mode applicability.
  If an IC or EAL includes an explicit reference to a technical specification, and the technical specification is not applicable because of operating mode, then that particular IC or EAL is also not applicable.

- **Example Emergency Action Level(s)** – these EALs are examples of conditions and indications that were considered to meet the criteria of the IC. These examples were not intended to be all encompassing, and some may not apply to a particular facility. Utilities should generally address each example EAL that applies to their site. If an example EAL does not apply because of its wording, e.g., specifies instrumentation not available at the site, the utility should identify other available means for entry into the IC. Ideally, the example EALs used will be unambiguous, expressed in site-specific nomenclature, and be readily discernible from control room instrumentation.

- **Basis** – provides information that explains the IC and example EALs. The bases are written to assist the personnel implementing the generic guidance into site-specific procedures. Site-specific deviations from the IC/EALs should be compared to the Basis for that IC to ensure that the fundamental intent of each IC/EAL is met. Some bases provide information intended to assist with establishing site-specific instrumentation values.

For Recognition Category F, basis information is presented in a format consistent with NEI 99-01 Rev 4. Tables 3 and 4. The presentation method shown for Fission Product Barrier Function Matrix was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. Other acceptable methods of achieving these goals which are currently in use include flow charts, block diagrams, and checklist tables. Utilities selecting these alternative need to ensure that all possible EAL combinations in the Fission Product Barrier Function Matrix are addressed in their presentation method.

For a number of Alerts, IC/EALs are chosen based on hazards which may cause damage to plant safety functions (i.e., tornadoes, hurricanes, FIRE in plant VITAL AREAs) or require additional help directly (control room evacuation) and thus increased monitoring of the plant is warranted. The symptom-based and barrier-based IC/EALs are sufficiently anticipatory to address the results of multiple failures, regardless of whether there is or is not a common cause. Declaration of the Alert will already result in the manning of the TSC for assistance and additional monitoring. Thus, direct escalation to the Site Area Emergency is unnecessary. Other Alerts, that have been specified, correspond to conditions which are consistent with the emergency class description.

The basis for declaring a Site Area Emergency and General Emergency is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.

With regard to the Hazards Recognition Category, the existence of a hazard that represents a potential degradation in the level of safety of the plant is the basis of NOUE classification. If the hazard results in VISIBLE DAMAGE to plant structures or equipment associated with safety systems or if system performance is affected, the event may be escalated to an Alert. The reference to "duration" or to "damage" to safety systems is intended only to size the event. Consequential damage from such hazards, if observed, would be the basis for escalation to Site Area Emergency or General Emergency, by entry to System Malfunction or Fission Product Barrier IC/EALs.

Basis Documents are attached as follows:

R- Abnormal Rad Levels / Radiological Effluent	R-1
C- Cold Shutdown / Refueling System Malfunction	C-1
E – Independent Fuel Storage Installation (USFSI)	E-1
F – Fission Product Barrier Degradation	F-1
H – Hazards	H-1
S- System Malfunction	S-1

6. DEFINITIONS

In the ICs and EALs, selected words are in uppercase print. These words are defined terms. Definitions are provided below.

A

ACTUATE: To put into operation; to move to action; commonly used to refer to automated, multi-faceted operations. "Actuate ECCS"

ADVERSARY: 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.

AFFECTING SAFE SHUTDOWN: event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable HOT or COLD

SHUTDOWN condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in HOT SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is not "AFFECTING SAFE SHUTDOWN."

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in COLD SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is "AFFECTING SAFE SHUTDOWN."

ALERT: Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guide exposure levels.

AVAILABLE: The state or condition of being ready and able to be used (placed into operation) to accomplish the stated (or implied) action or function. As applied to a system, this requires the operability of necessary support systems (electrical power supplies, cooling water, lubrication, etc.).

B

BOMB refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

C

CIVIL DISTURBANCE is a group of two or more persons violently protesting station operations or activities at the site.

CLOSE: To position a valve or damper so as to prevent flow of the process fluid. To make an electrical connection to supply power.

CONFINEMENT BOUNDARY is the barrier(s) between areas containing radioactive substances and the environment.

CONFIRM / CONFIRMATION: To validate, through visual observation or physical inspection, that an assumed condition is as expected or required, without taking action to alter the "As found" configuration.

CONTAINMENT CLOSURE is ~~defined by CL 1E, Containment Closure Checklist.~~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.

CONTIGUOUS: Being in actual contact; touching along a boundary or at a point

CONTROL: Take action, as necessary, to maintain the value of a specified parameter within applicable limits; to fix or adjust the time, amount, or rate of; to regulate or restrict.

D

DEVIATION: Instances where the guidance reference IC/EAL (99-01) differs in wording from proposed revision and is altered in intent, such that classification of event could be specifically different between guidance reference and licensee proposed EAL.

DIFFERENCE: Instances where the guidance reference IC/EAL (99-01) differ in wording but agree in meaning and intent.

DISCHARGE: Removal of a fluid/gas from a volume or system.

E

ENTER: To go into.

ESTABLISH: To perform actions necessary to meet a stated condition. "Establish communication with the Control Room."

EVACUATE: To remove the contents of; to remove personnel from an area.

EXCEEDS: To go or be beyond a stated or implied limit, measure, or degree.

EXIST: To have being with respect to understood limitations or conditions.

EXPLOSION is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

EXTORTION is an attempt to cause an action at the station by threat of force.

F

FAILURE: A state of inability to perform a normal function.

FAULTED: in a steam generator, the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE -is 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.

FISSION PRODUCT BARRIERS (FPB): Multiple physical barriers any of which, if maintained intact, precludes the release of significant amounts of radioactive fission

products to the environment. The FPBs are the Reactor Fuel Cladding (FC), Reactor Coolant System (RCS) and Containment (PC).

G

GENERAL EMERGENCY: Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed Protective Action Guide exposure levels offsite for more than the immediate site area.

H

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

I

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

INDICATE: To point out or point to; to display the value of a process variable; to be a sign or symbol.

INITIATE: The act of placing equipment or a system into service, either manually or automatically. Activation of a function or protective feature (i.e. initiate a manual trip).

INJECTION: The act of forcing a fluid into a volume or vessel.

INOPERABLE: Not able to perform its intended function.

INTRUSION / INTRUDER is a person(s) present in a specified area without authorization. Discovery of a BOMB in a specified area is indication of INTRUSION into that area by an ADVERSARY.

L

LOSS: Failure of operability or lack of access to.

LOWER: To become progressively less in size, amount, number, or intensity.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

M

MAINTAIN: Take action, as necessary, to keep the value of the specified parameter within the applicable limits.

MONITOR: Observe and evaluate at a frequency sufficient to remain apprised of the value, trend, and rate of change of the specified parameter.

N

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONs.

NOTIFY: To give notice of or report the occurrence of; to make known to, to inform specified personnel; to advise; to communicate; to contact; to relay.

O

OPEN: To position a valve or damper so as to allow flow of the process fluid. To break an electrical connection which removes a power supply from an electrical device. To make available for entry or passage by turning back, removing, or clearing away.

OPERABLE: Able to perform its intended function.

P

PERFORM: To carry out an action; to accomplish; to affect; to reach an objective.

PRIMARY SYSTEM: The pipes, valves, and other equipment which connect directly to the Reactor Vessel or reactor coolant system such that a reduction in Reactor Vessel pressure will effect a lowering in the steam or water being discharged through an unisolated break in the system.

PROTECTED AREA boundary is within the security isolation zone and is defined in the PBNP Safeguards Contingency Plan.

R

REMOVE: To change the location or position of.

REPORT: To describe as being in a specific state.

REQUIRE: To demand as necessary or essential.

RESTORE: Take the appropriate action required to return the value of an identified parameter to within the acceptable limits.

RISE: Describes an increase in a parameter as the result of an operator or automatic action. To become progressively greater in size, amount, number or intensity.

RUPTURED: In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

S

SABOTAGE is deliberate damage, misalignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may NOT meet the definition of SABOTAGE until this determination is made by security supervision.

SAFE PLANT SHUTDOWN: Hot or cold shutdown (reactor subcritical) with control of coolant inventory and decay heat removal.

SAFE SHUTDOWN SYSTEM: All cables, components, panels, power supplies, etc., necessary for a system to perform a safe shutdown function. Safe shutdown functions include: reactivity control, reactor coolant makeup, reactor heat removal, process system monitoring for variables necessary to control these functions and supporting functions such as component cooling, lubrication, etc., necessary for the operation of safe shutdown equipment.

SAMPLE: To perform an analysis on a specified media to determine its properties.

SHUTDOWN: To perform operations necessary to cause equipment to cease or suspend operation; to stop. "Shutdown unnecessary equipment."

SIGNIFICANT TRANSIENT is an UNPLANNED event involving one or more of the following: (1) automatic turbine runback >25% thermal reactor power, (2) electrical load rejection >25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations >10%.

SITE AREA EMERGENCY: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for 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.

SITE BOUNDARY: Per dose assessment methodology, the site boundary is approximately a one-mile radius around the site Protected Area.

STRIKE ACTION is a work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on PBNP. The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONs.

T

TRIP: To de-energize a pump or fan motor; to position a breaker so as to interrupt or prevent the flow of current in the associated circuit; to manually activate a semi-automatic feature.

U

UNAVAILABLE: Not able to perform its intended function.

UNCONTROLLED: An evolution lacking control but is not the result of an operator action.

UNPLANNED: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

UNUSUAL EVENT: Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

V

VALID: An indication, report, or condition is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator operability, the condition existence, or the report accuracy is removed. Implicit in this definition is the need for timely assessment.

VENT: To open an effluent (exhaust) flowpath from an enclosed volume; to reduce pressure in an enclosed volume.

VERIFY: To confirm a condition and take action to establish that condition if required. "Verify reactor trip."

VISIBLE DAMAGE is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREA is any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

Table ~~5-A-1~~R-0

Recognition Category AR

Abnormal Rad Levels / Radiological Effluent

INITIATING CONDITION MATRIX

~~NOUE~~		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
RAU1	Any UNPLANNED Release of Gaseous or Liquid Radio-activity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. *Op. Modes: All*	RAA1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the- Radiological Effluent Technical Specifications for 15 Minutes or Longer. *Op. Modes: All*	RAS1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 ~~mRem~~ mrem TEDE or 500 ~~mRem~~ mrem Thyroid CDE for the Actual or Projected Duration of the Release. *Op. Modes: All*	RAG1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 ~~mRem~~ mrem TEDE or 5000 ~~mRem~~ mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. *Op. Modes: All*
RAU2	Unexpected ~~Increase~~ Rise in Plant Radiation. *Op. Modes: All*	RAA3	Release of Radioactive Material or ~~Increases~~ Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown *Op. Modes: All*				
		RAA2	Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. *Op. Modes: All*				

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ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RAU1

Initiating Condition – ~~NOTIFICATION OF~~ UNUSUAL EVENT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer.

Operating Mode Applicability: All

~~Example Emergency Action Levels:~~Emergency Action Levels: (RU1.1 or RU1.2 or RU1.3~~ or 3 or 4 or 5~~)

RU1.1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.2. VALID reading on ~~one or more~~any of the following radiation monitors that exceeds the reading shown for 60 minutes or longer:

Table R-1 Radiation Monitors	
Monitor	Reading
RE-214 Auxiliary Building Vent Exhaust Gas	2.04E-04 µCi/cc
RE-315 Auxiliary Building Exhaust Low Range Gas	2.04E-04 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-04 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-04 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-03 µCi/cc*
2RE-309 Containment Purge Exhaust High Range Gas	4.18E-03 µCi/cc*
RE-221 Drumming Area Ventilation Gas	3.16E-04 µCi/cc
RE-325 Drumming Area Exhaust Low Range Gas	3.16E-04 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-04 µCi/cc
1(2)RE-229 Service Water Overboard	5.56E-05 µCi/cc
RE-230 Waste Water Effluent	2.06E-03 µCi/cc**

* with Unit 2 Containment purge or forced vent not occurring
** with Waste Water Effluent discharge not isolated
~~(site-specific list)~~

RU1.3. Confirmed sample analysis for gaseous or liquid releases indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times ~~(site-specific technical specifications)~~ RETS.

~~4. VALID reading on perimeter radiation monitoring system greater than 0.10 mR/hr above normal background sustained for 60 minutes or longer [for sites having telemetered perimeter monitors].~~

~~5. VALID indication on automatic real-time dose assessment capability greater than (site-specific value) for 60 minutes or longer [for sites having such capability].~~

Basis:

~~*Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.*~~

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. ~~Nuclear power plants~~PBNP incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the ~~Offsite Dose Calculation Manual (ODCM), and for plants that have not implemented Generic Letter 89-01, in the~~ Radiological Effluent Technical Specifications (RETS) and implemented as described in the PBNP Offsite Dose Calculation Manual (ODCM) [Ref. 1, 2]. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls. ~~Some sites may find it advantageous to address gaseous and liquid releases with separate initiating conditions and EALs.~~

The RETS multiples are specified in ICs ~~AU1~~ RU1 and ~~AA1~~ RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate. Releases should not be prorated or
averaged. For example, a release exceeding 4x RETS for 30 minutes does not meet the threshold for this IC.

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

~~EAL #1~~RU1.1 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the ~~Technical Specification~~ alarm setpoint established by the radioactivity discharge permit ~~limit~~ and releases are not terminated within 60 minutes. The appropriate detailed response to an effluent alarm is described in the PBNP RMS Alarm Set Point and Response Book (RMSARB) [Ref.3]. ~~This~~ These alarm setpoints may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the RETS. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

~~EAL #2~~RU1.~~1~~2 is intended for ~~licensees that have established~~ effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM

specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. ~~These monitor reading EALs should be determined using this methodology.~~ All of the monitors included in Table R-1 provide monitoring on non-routine effluent release pathways for which a discharge permit would not normally be prepared. The reading used for the classification threshold is two times the ODCM default set point for the applicable radiation monitor.

RE-214, -315, -317, and -319 are noble gas monitors used to monitor all gaseous effluent releases occurring through the Primary Auxiliary Building ventilation stack (common for Unit 1 and 2).

2RE-307 and -309 are noble gas monitors used to monitor all gaseous effluent releases occurring from the Unit 1 and 2 Letdown Gas Strippers which discharge through the Unit 2 Containment Purge ventilation stack. 2RE-307 and 309 were selected because the detectors have a higher operating range than the Letdown Gas Stripper Building discharge monitor (RE-224), which enables them to provide indication for the classification threshold. The reading used for the classification threshold is two times the ODCM default set point for RE-224. This reading assumes that a planned batch release from the Unit 2 Containment building (purge or forced ventilation release) is not occurring. Batch releases from the Unit 2 containment building are controlled using a discharge permit.

RE-221, -325, and -327 are noble gas monitors used to monitor all gaseous effluent releases occurring from the Drumming Area ventilation stack (common for Unit 1 and 2).

1(2) RE-229 is a liquid monitor used to monitor all liquid effluent releases occurring from Unit 1 and 2 which discharge through the Service Water system.

RE-230 is a liquid monitor used to monitor all liquid effluent releases occurring from Unit 1 and 2 which discharge through the Wastewater Effluent system.

~~EAL #3~~RU1.23 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in ~~river~~lake water systems, etc.

~~The 0.10 mR/hr value in EAL #4 is based on a release rate not exceeding 500 mrem per year, as provided in the ODCM / RETS, prorated over 8766 hours, multiplied by two, and rounded. (500 ÷ 8766 × 2 = 0.114). This is also the basis of the site specific value in EAL #5.~~

~~EALs #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the RETS and is used in calculating the alarm setpoints. EALs #4 and #5 are a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency. However, the fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #4 or EAL #5 is an indication of an uncontrolled release meeting the fundamental basis for this IC.~~

PBNP Basis Reference(s):

1. RETS – PBNP Technical Specification 5.5.1 and 5.5.4
2. Offsite Dose Calculation Manual (ODCM)
3. RMS Alarm Setpoint and Response Book (RMSASRB)
4. STPT 13.4, Radiation Monitoring System: Effluent Monitors

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ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

~~AU2~~RU2

Initiating Condition – ~~NOTIFICATION OF~~ UNUSUAL EVENT

Unexpected ~~Increase~~ Rise in Plant Radiation.

Operating Mode Applicability: All

~~Example Emergency Action Levels:~~Emergency Action Levels: (RU2.1 or RU2.2)

RU2.1. ~~a.~~ VALID ~~(site-specific)~~ indication of uncontrolled water level ~~decrease~~ lowering in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water as indicated by any of the following:

- Spent fuel pool low water level alarm setpoint
- Visual observation

AND

~~b.~~ Any UNPLANNED VALID ~~(site-specific)~~ ~~Direct~~ Area Radiation Monitor reading ~~increases~~ rises as indicated by:

- 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

RU2.2. Any UNPLANNED VALID ~~Direct~~ Area Radiation Monitor readings ~~increases~~ rises by a factor of 1000 over normal* levels ~~by a factor of 1000 over normal* levels.~~

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Basis:

This IC addresses increased radiation levels as a result of water level decreases above the ~~RPV~~ Reactor Vessel flange or events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and may represent a potential degradation in the level of safety of the plant~~.~~ [Ref. 1].

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via ~~EAL #1~~RU2.1 is appropriate given their potential for increased doses to plant staff. Classification as an ~~NO~~UE is warranted as a precursor to a more serious event. ~~Site-specific~~ Indications ~~may~~ include instrumentation such as water level and local area radiation monitors [Ref. 2], and personnel (e.g., refueling crew) reports. If available, security video cameras may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or decrease in refueling water storage tank level.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the reading on an area radiation monitor located near~~on~~ the refueling bridge may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Generally, increased radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss. For refueling events where the water level drops below the ~~RPV~~ Reactor Vessel flange classification would be via CU2. This event escalates to an Alert per IC ~~AA2~~ RA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Matrix for events in operating modes 1-4.

The low level alarm is actuated by LC-634, SFP Level Indicator at 62'-8" based on maintaining at least 6' of water on a withdrawn fuel assembly [Ref. 2].

~~EAL #2~~RU2.2 addresses UNPLANNED ~~increases~~ rises in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. This event escalates to an Alert per IC ~~AA3~~ RA3 if the increase in dose rates impedes personnel access necessary for safe operation.

~~*Normal levels can be considered as the highest reading (10 minute average) in the past twenty-four hours excluding the current peak value.~~

PBNP Basis Reference(s):

1. RMS Alarm Response Book (RMSARB)
2. DBD-13 Spent Fuel Pool Cooling and Filtration
3. STPT 13.1 Area Monitors

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

~~AA1~~RA1

Initiating Condition -- ALERT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer.

Operating Mode Applicability: All

~~Example Emergency Action Levels:~~Emergency Action Levels: (RA1.1 or RA1.2 or RA1.3~~ or 4 or 5~~)

RA1.1. VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

RA1.2. VALID reading on ~~one or more~~any of the following radiation monitors that exceeds the reading shown for 15 minutes or longer:

Table R-2 Radiation Monitors	
Monitor	Reading
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-02 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-02 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-01 µCi/cc*
2RE-309 Containment Purge Exhaust High Range	4.18E-01 µCi/cc*
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-02 µCi/cc

* with Unit 2 Containment purge or forced vent not occurring

~~(site-specific list)~~

RA1.3. Confirmed sample analysies for gaseous or liquid releases indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times ~~(site-specific technical specifications)~~ RETS.

~~4. VALID reading on perimeter radiation monitoring system greater than 10.0 mR/hr above normal background sustained for 15 minutes or longer [for sites having telemetered perimeter monitors].~~

~~5. VALID indication on automatic real-time dose assessment capability greater than (site-specific value) for 15 minutes or longer [for sites having such capability].~~

Basis:

~~*Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.*~~

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. ~~Nuclear power plants~~PBNP incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the ~~Offsite Dose Calculation Manual (ODCM), and for plants that have not implemented Generic Letter 89-01, in the~~ Radiological Effluent Technical Specifications (RETS) and implemented as described in the PBNP Offsite Dose Calculation Manual (ODCM) [Ref. 1, 2]. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls. ~~Some sites may find it advantageous to address gaseous and liquid releases with separate initiating conditions and EALs.~~

The RETS multiples -are specified in ICs ~~AU1~~-RU1 and ~~AA1~~-RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate [Ref. 2], the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate. Releases should not be prorated or averaged.

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

~~EAL #1~~RA1.1 addresses radioactivity releases that for whatever reason cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. ~~The values shown for each monitor [Ref. 4] under column "Alert" are approximately two hundred times the calculated alarm setpoints (ODCM release limits) as specified in the Radiation Monitoring Alarm Setpoint and Response Book (RMSASRB) [Ref. 3] choosing the highest values for variable conditions.~~ The appropriate detailed response to an effluent alarm is described in the PBNP RMS Alarm Set Point and Response Book (RMSASRB) [Ref. 3]. ~~This~~ These alarm setpoints may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the RETS. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

~~EAL #2 is similar to EAL #1, but is~~RA1.2 ~~intended to~~ addresses effluent or accident radiation monitors on non-routine release pathways (i.e., for which a discharge permit would not normally be prepared). The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. ~~These monitor reading EALs should be determined using this methodology.~~ All of the monitors included in Table R-2 provide monitoring on non-routine effluent release pathways for which a discharge permit would not normally be prepared. The reading used for the classification threshold is two hundred times the ODCM default set point for the applicable radiation monitor.

RE-317 and -319 are noble gas monitors used to monitor all gaseous effluent releases occurring through the Primary Auxiliary Building (PAB) ventilation stack (common for Unit 1 and 2). RE-317 and -319 were selected because the detectors have a higher operating range than the PAB

ventilation stack radiation monitor (RE-214), which enables them to provide indication for the classification threshold.

2RE-307 and -309 are noble gas monitors used to monitor all gaseous effluent releases occurring from the Unit 1 and 2 Letdown Gas Strippers which discharge through the Unit 2 Containment Purge ventilation stack. 2RE-307 and -309 were selected because the detectors have a higher operating range than the Letdown Gas Stripper Building discharge monitor (RE-224), which enables them to provide indication for the classification threshold. The reading used for the classification threshold is two hundred times the ODCM default set point for RE-224. This reading assumes that a planned batch release from the Unit 2 Containment building (purge or forced ventilation release) is not occurring. Batch releases from the Unit 2 containment building are controlled using a discharge permit.

RE-327 is a noble gas monitor used to monitor all gaseous effluent releases occurring from the Drumming Area ventilation stack (common for Unit 1 and 2). RE-327 was selected because the detector has a higher operating range than the Drumming Area ventilation stack radiation monitor (RE-221), which enables it to provide indication for the classification threshold. The reading used for the classification threshold is two hundred times the ODCM default set point for RE-221.

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 through RA1.1 or RA1.3 (e.g., sampling initiated due to entry into EAL RU1).

~~EAL #3~~RA1.3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in ~~river~~lake water systems, etc.

~~The 10.0 mR/hr value in EAL #4 is based on a release rate not exceeding 500 mrem per year, as provided in the ODCM / RETS, prorated over 8766 hours, multiplied by 200, and rounded. (500 ÷ 8766 × 200 = 11.4). This is also the basis of the site specific value in EAL #5.~~

~~EALs #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the RETS and is used in calculating the alarm setpoints. EALs #4 and #5 are a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency. However, the fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release. Exceeding EAL #4 or EAL #5 is an indication of an uncontrolled release meeting the fundamental basis for this IC.~~

Due to the uncertainty associated with meteorology, emergency implementing procedures ~~should~~ call for the timely performance of dose assessments using actual (real-time) meteorology in the event of a gaseous radioactivity release of this magnitude. The results of these assessments should be compared to the ICs ~~AS1~~-RS1 agbnd ~~AG1~~-RG1 to determine if the event classification should be escalated. ~~Contrary to the practices specified in revision 2 of this document, classification should not be delayed pending the results of these dose assessments.~~

PBNP Basis Reference(s):

1. RETS – PBNP Technical Specifications 5.5.1 and 5.5.4
2. Offsite Dose Calculation Manual (ODCM)
3. RMS Alarm Setpoint and Response Book (RMSASRB)

4. STPT 13.4, Radiation Monitoring System: Effluent Monitors

Initiating Condition – ALERT

Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

Operating Mode Applicability: All

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (RA2.1 or RA2.2)

RA2.1. A VALID ~~(site-specific)~~ high alarm or reading on ~~one or more~~ any of the following radiation monitors:

- o RE-105 SFP Area Low Range Radiation Monitor
- o RE-135 SFP Area High Range Radiation Monitor
- o RE 221 Drumming Area Ventilation Gas Monitor
- o RE 321 Drumming Area Exhaust Beta Particulate Monitor
- o RE 325 Drumming Area Exhaust Low Range Gas Monitor
- o 1(2) RE 102 El. 66' Containment Low Range Monitor
- o 1(2) RE-211 Containment Air Particulate Monitor
- o 1(2) RE 212 Containment Noble Gas Monitor

~~(site-specific monitors)~~

~~Refuel Floor Area Radiation Monitor~~
~~Fuel Handling Building Ventilation Monitor~~
~~Refueling Bridge Area Radiation Monitor~~

RA2.2. Water level ~~less than (site-specific) feet~~ LESS THAN 10 ft above an irradiated fuel assembly for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering.

Basis:

This IC addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent a degradation in the level of safety of the plant. These events escalate from IC ~~AU2~~ RU2 in that fuel activity has been released, or is anticipated due to fuel heatup. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in IC ~~E~~ AEU1.

~~EAL #1~~RA2.1 addresses radiation monitor indications [Ref. 1, 2, 3] of fuel uncovery and/or fuel damage. Increased readings on ventilation monitors may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the monitor due to

water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Application of these Initiating Conditions requires understanding of the actual radiological conditions present in the vicinity of the monitor. VALID high alarms indicated by the radiation monitors listed in RA2.1 may be indicative of a fuel handling accident and are, therefore, appropriate for this EAL [Ref. 1, 2, 3]. High alarm setpoint values and the appropriate detailed responses to radiation monitor high alarms are provided in the PBNP RMS Alarm Set Point and Response Book (RMSASRB) [Ref.1] Information Notice No. 90-08, *"KR-85 Hazards from Decayed Fuel"* ~~should be~~was considered in establishing radiation monitor EAL thresholds and there is no impact on this EAL.

Entry into ~~In EAL #2~~RA2.2 is based on ~~site-specific indications may include instrumentation such as water level and local area radiation monitors, and~~ personnel (e.g., refueling crew) reports. There is no site-specific water level instrumentation available that could be used for entry into this EAL. Water level lowering to less than 10 feet above an irradiated fuel assembly is indicative of conditions that will result in irradiated fuel uncovering while maintaining adequate radiation shielding to protect personnel in the area [Ref.7]. This EAL does not apply to planned activities that might require an irradiated fuel assembly to be raised to a level that is less than 10 feet from the surface of the water (e.g., maintenance or repair). ~~If available, security video cameras may allow remote observation. Depending on available level indication, the declaration threshold may need to be based on indications of water makeup rate or decrease in refueling water storage tank level.~~

Escalation, if appropriate, would occur via IC ~~AS1~~–RS1 or ~~AG1~~–RG1 or Emergency Director judgment.

PBNP Basis Reference(s):

1. RMS Alarm Setpoint and Response Book (RMSASRB)~~PBNP RMSASRB~~
2. AOP-8B Irradiated Fuel Handling Accident in Containment
3. AOP-8C Fuel Handling Accident in PAB
4. STPT 13.1 Area Monitors
5. STPT 13.2 Process Monitors
6. STP 13.4 Radiation Monitoring System: Effluent Monitors
7. DBD-05, Fuel Handling System

Initiating Condition -- ALERT

Release of Radioactive Material or ~~Increases~~ Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown

Operating Mode Applicability: All

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (RA3.1 or RA3.2)

RA3.1. VALID ~~(site-specific)~~ radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:

Control Room (RE-101)

OR

Central Alarm Station (by survey)

OR

Secondary Alarm Station (by survey)

~~(Site-specific) list~~

RA3.2. Any VALID ~~(site-specific)~~ radiation monitor readings GREATER THAN ~~<site-specific> values~~1 R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1).

Table H-1 Vital Areas
• 1(2) Containment Building • Primary Auxiliary Building • Turbine Building (by survey) • Control Building • Diesel Generator Building (by survey) • Gas Turbine Building (by survey) • Circ Water Pump House (by survey)

~~(Site-specific) list~~

Basis:

This IC addresses increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a

concern of this IC. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. For example, a dose rate of 15 mR/hr in the control room may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, an SAE or GE may be indicated by the fission product barrier matrix ICs.

At PBNP~~-multiple-unit-sites~~, the example EALs could result in declaration of an Alert at one unit due to a radioactivity release or radiation shine resulting from a major accident at the other unit. This is appropriate if the increase impairs operations at the operating unit.

This IC is not meant to apply to increases in the containment ~~dome~~-high range radiation monitors which normally indicate 1.0-2.5 R/hr, as these are events which are addressed in the fission product barrier matrix ICs. Nor is it intended to apply to anticipated temporary increases due to planned events (e.g., ~~incore-detector-movement,~~-radwaste container movement, incore detector movement, radiography, movement of large components, depleted resin transfers, etc.)

For RA3.1 areas requiring continuous occupancy include the Control Room, the central alarm station (CAS) and the secondary alarm station (SAS). The CAS and SAS have no installed radiation monitoring capability [Ref. 3], therefore entry into this EAL is based on radiation surveys in these areas. ~~.Areas-requiring-continuous-occupancy-includes-the-control-room-and,-as appropriate-to-the-site,-any-other-control-stations-that-are-manned-continuously,-such-as-a radwaste-control-room-or-a-central-security-alarm-station.~~-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. [Ref. 2, 3]

For RA3.2 areas requiring infrequent access, a valid radiation monitor reading greater than 1 R/hr would result in additional exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (e.g., current radiation protection and ALARA procedures, NMC administrative exposure limits, 10CFR20 limits, etc) and would impede necessary access.

As used here, *impede,* includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant.

The Turbine Building, Diesel Generator Building, Gas Turbine Building and Circ Water Pump House have no installed radiation monitor capability [Ref. 3], therefore entry into the EAL is based on radiation surveys in these areas.

Areas listed in Table H-1 were selected because they are areas or contiguous to areas requiring access to maintain plant safety functions.

~~Emergency-planners-developing-the-site-specific-lists-may-refer-to-the-site's-abnormal-operating procedures,-emergency-operating-procedures,-the-10-CFR-50-Appendix-R-analysis,-and/or,-the analyses-performed-in-response-to-Section-2.1.6b-of-NUREG-0578,-*"TMI-2-Lessons-Learned-Task Force-Status-Report-and-Short-term-Recommendations"*,-when-identifying-areas-containing-safe shutdown-equipment.-Do-not-use-the-dose-rates-postulated-in-the-NUREG-0578-analyses-as-a basis-for-the-radiation-monitor-readings-for-this-IC,-as-the-design-envelope-for-the-NUREG-0578 analyses-correspond-to-general-emergency-conditions.~~

PBNP Basis Reference(s):

1. GDC 19
2. NUREG-0737, "Clarification of TMI Action Plan Requirements", Section III.D.3
3. RMS Alarm Setpoint and Response Book (RMSASRB)

Initiating Condition -- SITE AREA EMERGENCY

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 ~~mR~~mrem TEDE or 500 ~~mR~~-mrem Thyroid CDE for the Actual or Projected Duration of the Release.

Operating Mode Applicability: All

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (RS1.1 or RS1.2 or RS1.3~~ or 4~~)

Note: *If dose assessment results are available at the time of declaration, the classification should be based on ~~EAL #2~~RS1.2 instead of ~~EAL #1~~RS1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

RS1.1. VALID reading on ~~one or more~~any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Table R-3 Radiation Monitors	
Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+00 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+00 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E-01 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E-01 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E-01 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-02 µCi/cc
1 S/G Safety Valve open	2.48E-02 µCi/cc
2 S/G Safety Valves open	1.24E-02 µCi/cc
3 S/G Safety Valves open	8.25E-03 µCi/cc
4 S/G Safety Valves open	6.20E-03 µCi/cc

~~(site-specific list)~~

RS1.2. Dose assessment using actual meteorology indicates doses GREATER THAN 100 ~~mR~~ mrem TEDE or 500 ~~mR~~-mrem thyroid CDE at or beyond the site boundary.

~~3. A VALID reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 100 mR/hr. [for sites having telemetered perimeter monitors]~~

4RS1.3. Field survey results indicate closed window dose rates exceeding 100 ~~mR~~mrem/hr expected to continue for more than one hour, at or beyond the site boundary;
~~or~~ **OR**

Aanalysies of field survey samples indicate thyroid CDE of 500 ~~mR~~ mrem for one hour of inhalation, at or beyond the site boundary.

Basis:

~~*Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.*~~

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed a small fraction of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone, e.g., fuel handling accident in spent fuel building.

The 100 mrem TEDE dose is set at 10% of the EPA PAG, while the 500 ~~mR~~ mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The ~~(site-specific)~~Table R-3 monitor list in ~~EAL #1~~ RS1.1 ~~should~~ includes monitors on all potential gaseous effluent release pathways. These monitors were selected because the detectors have an operating range that enables them to provide indication for the classification threshold.

1(2) RE-307 and –309 are noble gas monitors used to monitor all releases occurring from the Unit and 2 Containment purge ventilation stacks. RE-317 and –319 are noble gas monitors used to monitor all releases occurring from the Drumming Area ventilation stack (common for Unit 1 and 2). 1(2)RE-231 and 1(2)RE-232 are process monitors used to monitor all releases occurring through the Unit 1 and 2 atmospheric steam dump and S/G safety valves.

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 calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facility's emergency planning zone.~~

The monitor reading thresholds for RS1.1 were determined by multiplying the monitor readings in EAL RG1.1 Table R-4 by 0.1 to determine the monitor reading thresholds corresponding to 500 mrem Thyroid CDE at or beyond the site boundary (1 mile downwind), which is the more limiting dose value for these calculations [Ref.1]. The inputs used for the calculations in Reference 1 are described in the Basis section for EAL RG1.
~~The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. The meteorology and source term (noble gases, particulates, and halogens) used should be the same as those used for determining the monitor reading EALs in ICs AU1 and AA1. This protocol will maintain intervals between the EALs for the four classifications. Since doses are generally not monitored in real time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site boundary (or beyond) dose of 100 mR/hour whole body or 500 mR/hour thyroid, whichever is more limiting (as was done for EALs #3 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.~~

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures ~~should~~ call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.
~~Contrary to the practices specified in revision 2 of this document, classification should not be delayed pending the results of these dose assessments.~~

PBNP Basis Reference(s):

1. Effluent Monitor Classification Thresholds WEDAP Basis Document, NPC 2004-00731
2. EPIP 1.3 Dose Assessment and Protective Action Recommendations
3. FSAR Table 2.6-3 Stability Index Distribution
4. FSAR Table 2.6-4 Site Atmospheric Stability Analysis Annual Average 13 month Data
5. FSAR Figure 2.6-2 Stability Class Distribution in Percent of Total Observed
6. USEPA 400-R-92-001, -Manual of Protective Action Guidelines and Protective Actions for Nuclear Incidents
7. DBD-T-46 Section 3.1, Station Blackout

Initiating Condition -- GENERAL EMERGENCY

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 ~~mR~~-mrem TEDE or 5000 ~~mR~~-mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability: All

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (RG1.1 or RG1.2 or RG1.3~~ or 4~~)

Note: *If dose assessment results are available at the time of declaration, the classification should be based on ~~EAL #2~~RG1.2 instead of ~~EAL #1~~RG1.1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

RG1.1. VALID reading on ~~one or more~~any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Table R-4 Radiation Monitors	
Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+01 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+01 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E+00 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E+00 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E+00 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-01 µCi/cc
1 S/G Safety Valve open	2.48E-01 µCi/cc
2 S/G Safety Valves open	1.24E-01 µCi/cc
3 S/G Safety Valves open	8.25E-02 µCi/cc
4 S/G Safety Valves open	6.20E-02 µCi/cc

~~(site-specific list)~~

RG1.2. Dose assessment using actual meteorology indicates doses GREATER THAN 1000 ~~mR~~ mrem TEDE or 5000 ~~mR~~-mrem thyroid CDE at or beyond the site boundary.

~~3. A VALID reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 1000 mR/hr. [for sites having telemetered perimeter monitors]~~

4RG1.3. Field survey results indicate closed window dose rates exceeding 1000 ~~mR~~mrem/hr expected to continue for more than one hour, at or beyond site boundary.~~; or a~~

OR

Analysies of field survey samples indicate thyroid CDE of 5000 ~~mR~~-mrem for one hour of inhalation, at or beyond site boundary.

Basis:

~~*Refer to Appendix A for a detailed basis of the radiological effluent IC/EALs.*~~

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The ~~(site-specific)~~Table R-4 monitor list in ~~EAL #1~~RG1.1 ~~should~~ includes monitors on all potential gaseous effluent release pathways. These monitors were selected because the detectors have a high enough operating range that enables them to provide indication for the classification threshold.

1(2) RE-307 and –309 are noble gas monitors used to monitor all releases occurring from the Unit 1 and 2 Containment purge ventilation stacks. RE-317 and –319 are noble gas monitors used to monitor all releases occurring through Primary Auxiliary Building (PAB) ventilation stack 9common for Unit 1 and 2). RE-327 is a noble gas monitor used to monitor all releases occurring from the Drumming Area ventilation stack (common for Unit 1 and 2). 1(2)RE-231 and 1(2)RE-232 are process monitors used to monitor all releases occurring through the Unit 1 and 2 atmospheric steam dump and S/G safety valves.

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 calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facilities emergency planning zone.~~

The monitor reading thresholds for RG1.1 were determined using the Wisconsin Electric Dose Assessment Program (WEDAP) computer code and annual average meteorology [Ref.1]. The monitor readings determined in Reference 1 were back calculated from a dose value of 5000 mrem Thyroid CDE at or beyond the site boundary (1 mile downwind), which is the more limiting dose value for these calculations. The inputs used for the calculations in Reference 1 are as follows:

Wind direction: from 30 degrees (NNE to SSW)
Basis: FSAR Table 2.6-4 Annual average meteorology
Wind speed: 10 mph
Basis: FSAR Table 2.6-4 Annual average meteorology for stability class D (8-12 mph)
Stability class: D
Basis: FSAR Tables 2.6-2 and 2.6-3 Annual average meteorology - Stability index distribution and Stability Class Distribution in Percent of Total Observed

and WEDAP default stability class
Time after shutdown: 4 hrs.
Basis: FSAR Appendix A-1 and DBD-T-46 Section 3.1
Station Blackout Coping Time (time to core damage)
Release duration: 4 hrs. (default)
Containment purge vs. forced vent: 1 purge fan
Releases filtered except for 1(2) RE 231/232 Steam Line Monitors
No lake breeze effect
No precipitation
Building Wake Effect (default)
Source term: LOCA/Gap release inside containment except for 1(2) RE 231/232 Steam line Monitors which were based on Gap release/SGTR
No containment sprays
SGTR release path assumes SG water level <29% narrow range
Calculation results for RE 231were used for monitor reading threshold for RE 232 (similar monitor and location)

~~The monitor reading EALs should be determined using a dose assessment method that backcalculates from the dose values specified in the IC. The meteorology and source term (noble gases, particulates, and halogens) used should be the same as those used for determining the monitor reading EALs in ICs AU1 and AA1. This protocol will maintain intervals between the EALs for the four classifications. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site boundary (or beyond) dose of 1000 mR/hour whole body or 5000 mR/hour thyroid, whichever is more limiting (as was done for EALs #3 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.~~

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures ~~should~~ call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.
~~Contrary to the practices specified in revision 2 of this document, classification should not be delayed pending the results of these dose assessments.~~

PBNP Basis Reference(s):

1. Effluent Monitor Classification Thresholds WEDAP Basis Document, NPC 2004-00731
2. EPIP 1.3 Dose Assessment and Protective Action Recommendations
3. FSAR Table 2.6-3 Stability Index Distribution
4. FSAR Table 2.6-4 Site Atmospheric Stability Analysis Annual Average 13 month Data
5. FSAR Figure 2.6-2 Stability Class Distribution in Percent of Total Observed
6. UDSEPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
7. DBD-T-46 Section 3.1, Station Blackout
8. FSAR Appendix A-1, Station Blackout

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Table C-0
Recognition Category C
Cold Shutdown/Refueling System Malfunction
INITIATING CONDITION MATRIX

~~NOUUE~~	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
CU1 RCS Leakage. *Op. Mode: Cold Shutdown*	**CA1** Loss of RCS Inventory. *Op. Modes: Cold Shutdown*	**CS1** Loss of ~~RPV~~Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability. *Op. Modes: Cold Shutdown*	**CG1** Loss of ~~RPV~~Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the ~~RPV~~Reactor Vessel. Op. Modes: Cold Shutdown, Refueling
CU2 UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the ~~RPV~~Reactor Vessel *Op. Mode: Refueling*	**CA2** Loss of ~~RPV~~Reactor Vessel Inventory with Irradiated Fuel in the ~~RPV~~Reactor Vessel. *Op. Modes: Refueling*	**CS2** Loss of ~~RPV~~Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the ~~RPV~~Reactor Vessel. Op. Modes: Refueling	
CU3 Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes. *Op. Modes: Cold Shutdown, Refueling*	**CA3** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Cold Shutdown, Refueling, Defueled*		
CU4 UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the ~~RPV~~Reactor Vessel. *OP. Modes: Cold Shutdown, Refueling*	**CA4** Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the ~~RPV~~Reactor Vessel. *Op. Modes: Cold Shutdown, Refueling*		
~~**CU5** Fuel Clad Degradation. *Op. Modes: Cold Shutdown, Refueling*~~			
CU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities. *Op. Modes: Cold Shutdown, Refueling*			
CU7 UNPLANNED Loss of Required DC Power for Greater than 15 Minutes. *Op. Modes: Cold Shutdown, Refueling*			

CU8 Inadvertent Criticality.
Op Modes:, Cold Shutdown, Refueling

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SYSTEM MALFUNCTION

CU1

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

RCS Leakage.

Operating Mode Applicability: Cold Shutdown

~~Example~~ Emergency Action Levels: (CU1.1 or CU1.2)

CU1.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm.

CU1.2. Identified leakage GREATER THAN 25 gpm.

Basis:

This IC is included as a ~~NOUE~~ UE because it is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is sufficiently large to be observable via normally installed instrumentation (e.g., Pressurizer level, RCS loop level instrumentation, etc...~~) or reduced inventory instrumentation such as level hose indication~~. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). OI 55 provides instructions for calculating primary system leak rate by water inventory balances~~ for off normal events and for operations troubleshooting~~. The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Prolonged loss of RCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of RCS Inventory with Irradiated Fuel in the ~~RPV~~Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the ~~RPV~~Reactor Vessel).

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and RCS inventory and level monitoring means such as Pressurizer level indication and makeup volume control tank levels are normally available. In the refueling mode the RCS is not intact and ~~RPV~~Reactor Vessel level and inventory are monitored by different means.

~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

~~OP4A, Filling and Venting Reactor Coolant System, Step 5.6~~

1. TS 3.4.13, RCS Operational Leakage limits
2. OI 55, Primary Leak Rate Calculation
3. OM 3.19, Reactor Coolant System Leakage Determination

CU2

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the ~~RPV~~Reactor Vessel.

Operating Mode Applicability: Refueling

~~Example Emergency Action Levels:~~Emergency Action Levels: (CU2.1 or CU2.2)

CU2.1. UNPLANNED RCS level ~~decrease~~ lowering below the ~~RPV~~ Reactor Vessel flange (89.1%) for ≥ GREATER THAN OR EQUAL TO 15 minutes

CU2.2. ~~a.~~ Loss of ~~RPV~~ Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank~~{site-specific} sump and tank~~ level ~~increase~~rise

AND

~~b.~~ ~~RPV~~ Reactor Vessel level cannot be monitored

Basis:

This IC is included as an ~~NOUE~~ UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Refueling evolutions that decrease RCS water level below the ~~RPV~~ Reactor Vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the ~~RPV~~ Reactor Vessel flange warrants declaration of an ~~NOUE~~ Unusual Event due to the reduced RCS inventory that is available to keep the core covered. The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists. Continued loss of RCS Inventory will result in escalation to the Alert level via either IC CA2 (Loss of ~~RPV~~ Reactor Vessel Inventory with Irradiated Fuel in the ~~RPV~~Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the ~~RPV~~Reactor Vessel).

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and ~~RPV~~ Reactor Vessel level and inventory are monitored by different means.

In the refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of ~~RPV~~Reactor Vessel level indication will normally be used~~installed~~ (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that ~~RPV~~Reactor Vessel inventory loss was occurring by observing ~~sump~~ Containment Sump A and ~~tank~~ Waste Holdup Tank level changes [Ref. 1, 2]. ~~Sump~~ Sump and tank level ~~increase~~rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. OI 55 [Ref. 4] provides instructions for calculating primary system

leak rate by water inventory balances. Containment Sump A is equipped with a high level alarm (80%). Escalation to Alert would be via either CA2 or RCS heatup via CA4.

~~EAL 1~~CU2.1 involves a decrease in RCS level below the top of the ~~RPV~~Reactor Vessel flange that continues for 15 minutes due to an UNPLANNED event. The Reactor Vessel flange is at elevation 40 ft 8 in., which is 89% on LI-447/447A or 89 in. on LI-447B [Ref. 3]. This EAL is not applicable to decreases in flooded reactor cavity level (covered by ~~AU2-EAL1~~RU2.1) until such time as the level decreases to the level of the vessel flange. ~~For BWRs, if RPV level continues to decrease and reaches the Low-Low ECCS Actuation Setpoint then escalation to CA2 would be appropriate. For PWRs, i~~If ~~RPV~~Reactor Vessel level continues to decrease and reaches the Bottom ID of the RCS Loop, (33 ft 2-7/8 in. elev. or 0%/0 in.), then escalation to CA2 would be appropriate. Note that the Bottom ID of the RCS Loop Setpoint ~~should be the level equal~~corresponds to the bottom of the ~~RPV~~Reactor Vessel loop penetration (not the low point of the loop).

~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

1. ARB C01 B 1-4, UNIT 1 CONTAINMENT SUMP A LEVEL HIGH
2. STPT 12.1, Waste Disposal System
3. OP 4D Part 3, Draining the Reactor Cavity and Reactor Coolant System, Table 1
4. OI 55, Primary Leak Rate Calculation

SYSTEM MALFUNCTION

CU3

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses for G~~reater Tha~~REATER THANn 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

~~Example~~ Emergency Action Level:

CU3.1. ~~a.~~ Loss of all **offsite** power to ~~(site-specific) transformers~~ both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN~~greater than~~ 15 minutes.

AND

~~b.~~ At least ~~(site-specific)[number of~~ 1 emergency generator~~s~~ ~~are~~is supplying power to an emergency bus.~~ses.~~

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The 4160 VAC system includes two safety-related (essential) buses per unit, 1(2)-A05 (A train) and 1(2)-A06 (B train). Offsite power from the 345 KVAC system is stepped down through the 13.8 KVAC system to the Low Voltage Station Auxiliary Transformer (LVSAT) 1(2)-X04. The LVSATs provide power to 4160 VAC switching buses 1(2)-A03 and 1(2)-A04, which in turn provide power to safety-related buses 1(2)-A05 and 1(2)-A06.

During emergency or abnormal situations, the 4160 VAC system is supplied by emergency diesel generators (G01 through G04) or the gas turbine generator (G05). Following a loss of power, ECA 0.0 provides guidance to restore power to any 4160 VAC safety-related bus. For the purpose of classification under this EAL, offsite power sources include any of the following:

- 345 KVAC system supplying power to the 13.8 KVAC system and the unit LVSATs
- Cross-tying with the opposite unit power supply
- Backfeeding power to the UATs through the 19 KVAC system and the main step-up transformer X-01. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source.

~~Plants that have~~PBNP has the capability to cross-tie AC power from ~~a companion~~the other unit ~~may~~ and therefore takes credit for the redundant power source ~~in the associated EAL~~ for this IC. However, the ~~i~~Inability to effect the cross-tie within 15 minutes warrants declaring a ~~N~~OUE.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
2. DBD-18, 13.8 KVAC System, Section 3.3.0
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A U1, Main Power Transformer Backfeed

SYSTEM MALFUNCTION

CU4

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the ~~RPV~~Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

~~Example Emergency Action Levels:~~Emergency Action Levels: (CU4.1 or CU4.2)

CU4.1. An UNPLANNED event results in RCS temperature exceeding ~~the Technical Specification cold shutdown temperature limit~~]200°F

CU4.2. Loss of all RCS temperature and ~~RPV~~Reactor Vessel level indication for ~~>~~GREATER THAN 15 minutes.

Basis:

This IC is included as an ~~NOUE~~UE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered. In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power. Entry into the refueling mode procedurally may not occur for typically 100 hours ~~{site-specific}~~ or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). In addition, the operators should be able to monitor RCS temperature and ~~RPV~~Reactor Vessel level so that escalation to the alert level via CA4 or CA1 will occur if required.

During refueling the level in the ~~RPV~~Reactor Vessel will normally be maintained above the ~~RPV~~Reactor Vessel flange. Refueling evolutions that decrease water level below the ~~RPV~~Reactor Vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid ~~increase~~rises in RCS/~~RPV~~Reactor Vessel temperatures depending on the time since shutdown. Escalation to the Alert level is via CA4. ~~is provided should an UNPLANNED event result in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for greater than 30 minutes with CONTAINMENT CLOSURE not established.\~~

Unlike the cold shutdown mode, normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of ~~RPV~~Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold

shutdown orf refueling modes, ~~EAL 2~~CU4.2 would result in declaration of an ~~NOUE~~ Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via CA2 based on an inventory loss or CA4 based on exceeding its temperature criteria (200°F) [Ref. 1].

Reactor Vessel water level is normally monitored using the following instruments [Ref. 2, 4]:

- LT-494, 495 (RVLIS) Reactor Vessel WR Water Level (0 - 125 ft)
- LT-496, 497 (RVLIS) Reactor Vessel NR Water Level (0 - 45 ft)
- LT-447, 447A Reduced Inventory RV Water Level (0 - 100 in.)

LT-494, 495, 496, 497 provide appropriate level signals to allow the Control Room operator to monitor the associated parameter during design basis accidents. These instruments are only to be used for trend information and are not used for definitive Reactor Vessel level indication when draining the reactor cavity or RCS.

LT-447, 447A provide Reactor Vessel water level indication in the Control Room during reduced RCS inventory condition. They are calibrated to indicate level from approximately 10 in. above the vessel flange to a level approximately 90 inches below the vessel flange (bottom of the hot leg). The instrument scale range is 0 -100% which corresponds to 0 -100 in. The readout can be displayed on the Plant Process Computer System (PPCS) at the operator's desk and on the control board with a readout of ±0.1 inches while at mid-loop. This provides the ability to read the level from the bottom of the reactor coolant pipe (0% or 33 ft 2-7/8 in. elev.) to well above the Reactor Vessel head flange (89.1% or 40 ft 8 in. elev.).

Local indicator LI-447B, RC Reduced Inventory Level Indicator (0 -100 in.), also provides Reactor Vessel water level indication. 0 in. on LI-447B corresponds to 0% on LI-447/447A.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include [Ref. 2, 3]:

- Tavg instrumentation
- TE-1 thru 39, Core Exit Thermocouples (50 - 1600°F)
- TE-450A, 450C, 451A, 451C, Cold Leg Loop WR Temperature (50 - 750°F)
- TE-450B, 450D, 451B, 451D, Hot Leg Loop WR Temperature (50 - 750°F)
- TE-630, RHR Inlet Temperature

Loop WR Temperature Elements provide wide range RCS temperature signals for monitoring heatup and cooldown, and unusual events such as natural circulation, where the RTDs in the loop bypass lines do not provide meaningful temperature signals. Only Loop B Cold Leg WR temperature indication uncertainties have been analyzed to ensure the PTLR limits are maintained. Loop B cold leg WR temperature channel is used when on RHR. TC_{AVG} is preferred thermocouple indication and provides an average of all thirty-nine thermocouples and the trend recorders can look at one of eight or an average of all eight thermocouples.

The Emergency Director must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an

imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

1. Technical Specifications Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
2. DBD-09, Reactor Coolant System, Rev. 6, Sections 3.23.1 to 3.23.4, 3.24.1, 3.24.2, 3.25.1 and 3.25.2
3. OI 105, RCS Heatup/Cooldown Plotting, Rev. 9
4. OP 4D Part 3, Draining the Reactor Cavity and Reactor Coolant System, Table 1, Rev. 16

~~SYSTEM MALFUNCTION~~

~~CU5~~

~~Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT~~

~~Fuel Clad Degradation.~~

~~Operating Mode Applicability:~~ ~~Cold Shutdown~~
~~Refueling~~

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

~~1. (Site-specific) radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits.~~

~~2. (Site-specific) coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.~~

~~Basis:~~

~~This IC is included as a NOUE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. EAL #1 addresses site-specific radiation monitor readings that provide indication of fuel clad integrity. EAL #2 addresses coolant samples exceeding coolant technical specifications for iodine spike.~~

SYSTEM MALFUNCTION

CU6

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Cold Shutdown
Refueling

~~Example Emergency Action Levels:~~Emergency Action Levels: (CU6.1 or CU6.2)

CU6.1. Loss of all ~~(site-specific list)~~ Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Plant Public Address System • Security Radio • Commercial Phone System • Portable radios • Sound power phones

CU6.2. Loss of all ~~(site-specific list)~~ Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• Emergency Notification System (ENS) • Health Physics Network (HPN) • Operations Control Counterpart Link (OCCL) • Management Counterpart Link (MCL) • Protective Measures Counterpart Link (PMCL) • Reactor Safety Counterpart Link (RSCL) • Nuclear Accident Reporting System (NARS) • Commercial Phone System • General Telephone Lines • Manitowoc County Sheriff's Department Radio

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

~~Site-specific list for~~Table C-1 onsite communications loss [Ref. 1, 2]~~must~~ encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system and radios / walkie talkies).

~~Site-specific list for~~Table C-2 offsite communications loss [Ref. 1, 2]~~must~~ encompasses the loss of all means of communications with offsite authorities. This ~~should~~ includes the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

PBNP Basis Reference(s):

1. EPMP 2.1, Testing of Communications Equipment
2. EPMP 2.1A, Monthly Communications Test

SYSTEM MALFUNCTION

CU7

Initiating Condition – ~~NOTIFICATION OF~~ UNUSUAL EVENT

UNPLANNED Loss of Required DC Power for ~~Greater than~~GREATER THAN 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

~~Example Emergency~~Emergency Action Level:

~~1. a.~~ CU7.1 -UNPLANNED Loss of v~~V~~ital DC power to required DC busses based on ~~(site-specific)~~ bus voltage indications LESS THAN 115 VDC~~.~~

AND

~~b.~~ Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

UNPLANNED is included in this IC and EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely plants will perform maintenance on a Train related basis during shutdown periods. It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per CA4 "Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the ~~RPV~~Reactor Vessel."

~~(Site-specific)~~LESS THAN 115 VDC bus voltage ~~should be~~is based on the minimum bus voltage necessary for the operation of safety related equipment [Ref. 1, 2]. ~~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. Typically the value for the entire battery set is approximately 105 VDC. For a 60 cell string of batteries the cell voltage 1.75 Volts per cell. For a 58 string battery set the minimum voltage is typically 1.81 Volts per cell.~~

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

The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses.

PBNP Basis Reference(s):

1. FSAR Section 8.7
2. 0-SOP-DC-001/2/3/4 125 VDC System, Bus D-01, D-02, D-03, D-04 and Components

SYSTEM MALFUNCTION

CU8

Initiating Condition – ~~NOTIFICATION OF~~ UNUSUAL EVENT

Inadvertent Criticality.

Operating Mode Applicability: Cold Shutdown
Refueling

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

~~1. An UNPLANNED extended positive period observed on nuclear instrumentation~~

CU8.~~2~~1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events and inadvertent dilution events. This IC indicates a potential degradation of the level of safety of the plant, warranting an ~~NOUE~~ Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated) which are addressed in the companion IC SU8.

This condition can be identified using ~~period monitors/~~the startup rate monitor. The terms ~~"extended" and~~ "sustained" ~~are~~ is used in order to allow exclusion of expected short term positive ~~periods/~~startup rates from planned fuel bundle or control rod movements during core alterations ~~for PWRs and BWRs~~. These short term positive ~~periods/~~startup rates are the result of the ~~increase~~rise in neutron population due to subcritical multiplication.

This condition can be identified using startup rate monitors (NI-31D/32D - Source Range Startup Rate, and NI-35D/36D - Intermediate Range Startup Rate) [Ref. 1].

Escalation would be by Emergency Director Judgment.

PBNP Basis Reference(s):

1. OP 1B, Reactor Startup, Rev 50~~-~~,Step 5.1 and 5.17.15

Initiating Condition – ALERT

Loss of RCS Inventory.

Operating Mode Applicability: Cold Shutdown

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (CA1.1 or CA1.2)

CA1.1. Loss of RCS inventory as indicated by ~~RPV~~Reactor Vessel level LESS THAN 6% on LI-447 / LI-447A ~~{site-specific level}.~~
~~(low-low ECCS actuation setpoint) (BWR)~~
~~(bottom ID of the RCS loop) (PWR)~~

CA1.2. ~~a.~~ Loss of RCS inventory as indicated by unexplained ~~{site-specific} sump~~ Containment Sump A ~~and~~ or Waste Holdup Ttank level ~~increase~~rise

AND

~~b.~~ RCS level cannot be monitored for ~~>~~ GREATER THAN 15 minutes

Basis:

These ~~example~~ EALs serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further ~~RPV~~Reactor Vessel level decrease and potential core uncovery. 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. ~~The BWR Low-Low ECCS Actuation Setpoint was chosen because it is a standard setpoint at which all available injection systems automatically start.~~ The ~~PWR~~ 6 inch level ~~Bottom ID~~ of the RCS Loop ~~Setpoint~~ was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred [Ref. 1]. The 6 inch level is above the CS2 setpoint of 0, ~~Bottom ID of the RCS Loop Setpoint should be is~~ the level equal to the bottom of the ~~RPV~~Reactor Vessel loop penetration ~~(not the low point of the loop)~~. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours ~~{site-specific}~~ or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold

shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the cold shutdown mode, normal RCS level and ~~RPV~~Reactor Vessel level instrumentation systems will normally be available. During preparations for refueling, reactor vessel level indication may not be available. In this instance, classification would be completed under CA1.2 for sump and tank level indications. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that ~~RPV~~Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level ~~increase~~rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage [Ref. 1, 2]. The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency EAL duration. The 15-minute duration allows CA1 to be an effective precursor to CS1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS1 basis. Therefore this EAL meets the definition for an Alert emergency.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and ~~RPV~~Reactor Vessel level and inventory are monitored by different means.

If ~~RPV~~Reactor Vessel level continues to decrease then escalation to Site Area will be via CS1 (Loss of Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the ~~RPV~~Reactor Vessel).

~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

1. OP 4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
2. OM 3.19, REACTOR COOLANT SYSTEM LEAKAGE DETERMINATION

SYSTEM MALFUNCTION

CA2

Initiating Condition – ALERT

Loss of ~~RPV~~Reactor Vessel Inventory with Irradiated Fuel in the ~~RPV~~Reactor Vessel.

Operating Mode Applicability: Refueling

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (CA2.1 or CA2.2)

CA2.1. Loss of ~~RPV~~Reactor Vessel inventory as indicated by ~~RPV~~Reactor Vessel level LESS THAN 6% on LI-447 / LI-447A ~~{site-specific level}~~.

CA2.2. ~~a.~~ Loss of ~~RPV~~Reactor Vessel inventory as indicated by unexplained Containment Sump A~~{site-specific} sump~~ ~~and~~ or Waste Holdup T~~t~~ank level ~~increase~~rise

AND

~~b.~~ ~~RPV~~Reactor Vessel level cannot be monitored for >GREATER THAN 15 minutes

Basis:

These ~~example~~ EALs serve as precursors to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further ~~RPV~~Reactor Vessel level decrease and potential core uncovery. The LI-447 / LI-447A threshold corresponds to the 6 inches above the bottom inside diameter of the RCS loop [Ref. 1]. The 6 inch level of the RCS Loop was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. This condition will result in a minimum classification of Alert. The 6 inch level is above the CS2 setpoint of 0, the level equal to the bottom of the Reactor Vessel loop penetration.~~The BWR Low-Low ECCS Actuation Setpoint was chosen because it is a standard setpoint at which all available injection systems automatically start. The Bottom ID of the RCS Loop Setpoint was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems may occur.~~ The Bottom ID of the RCS Loop Setpoint ~~should be~~is the level equal to the bottom of the ~~RPV~~Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours ~~{site-specific}~~ or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold

shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the refueling mode, normal means of ~~RPV~~Reactor Vessel level indication may not be available. Redundant means of ~~RPV~~Reactor Vessel level indication will be normally used~~installed~~ (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that ~~RPV~~Reactor Vessel inventory loss was occurring by observing sump and tank level changes [Ref. 1, 2]. Sump and tank level ~~increase~~rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS2 Site Area Emergency EAL duration. The 15-minute duration allows CA2 to be an effective precursor to CS2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS2 basis. Therefore this EAL meets the definition for an Alert.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and ~~RPV~~Reactor Vessel level and inventory are monitored by different means.

If ~~RPV~~Reactor Vessel level continues to decrease then escalation to Site Area will be via CS1 (Loss of Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the ~~RPV~~Reactor Vessel).

~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

1. OP 4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
2. OM 3.19, REACTOR COOLANT SYSTEM LEAKAGE DETERMINATION

Initiating Condition -- ALERT

Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability: Cold Shutdown
Refueling
Defueled

~~Example Emergency~~Emergency Action Level:

CA3.1. ~~a.~~ Loss of all offsite power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06

~~power to (site-specific) transformers.~~

AND

~~b.~~ Failure of ~~(site-specific)~~all emergency generators to supply power to emergency busses.

AND

~~c.~~ Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Escalating to Site Area Emergency IC SS1, if appropriate, is by Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

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 essential busses. Even though an essential 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.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
2. DBD-18, 13.8 KVAC System, Section 3.3.0
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems

SYSTEM MALFUNCTION

CA4

Initiating Condition -- ALERT

Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the ~~RPV~~Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (EAL CA4.1 or CA4.2 or CA4.3)

CA4.1. With CONTAINMENT CLOSURE and RCS integrity not established an UNPLANNED event results in RCS temperature exceeding ~~the Technical Specification cold shutdown temperature limit~~200 degrees F.

CA4.2. With CONTAINMENT CLOSURE established and RCS integrity not established or RCS inventory reduced an UNPLANNED event results in RCS temperature exceeding ~~the Technical Specification cold shutdown temperature limit~~200 degrees F for ~~greater than~~GREATER THAN 20 minutes[1].

CA4.3. An UNPLANNED event results in RCS temperature exceeding ~~the Technical Specification cold shutdown temperature limit~~200 degrees F for GREATER THAN~~greater than~~ 60 minutes[1] or results in an RCS pressure ~~increase~~rise of GREATER THAN~~greater than {site specific}~~10 psig.

Basis:

~~EAL 1~~CA4.1 addresses complete loss of functions required for core cooling during refueling and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). No delay time is allowed for ~~EAL1~~ CA4.1 because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

~~EAL 2~~CA4.2 addresses the complete loss of functions required for core cooling for >GREATER THAN 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is reduced (e.g., mid loop operation~~ in PWRs~~). As in ~~EAL 1~~CA4.1, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed later in this basis) and is believed to be conservative given that a low pressure Containment barrier to fission product release is established. Note 1 indicates that ~~EAL 2~~CA4.2 is

[1]Note: if an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.

not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 20 minute time frame.

~~EAL 3~~CA4.3 addresses complete loss of functions required for core cooling for >GREATER THAN 60 minutes during refueling and cold shutdown modes when RCS integrity is established. As in ~~EAL 1~~CA4.1 and ~~2~~CA4.2, RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this EAL is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The ~~{site-specific}~~10 psig pressure ~~increase~~rise covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. Pressure indicators PT-420 (RCS loop pressure) and PT-493 (pressurizer pressure) have a range of 0 to 1000 psig and are capable of measuring pressure to less than 10 psig [Ref. 6]. ~~The RCS pressure setpoint chosen should be 10 psig or the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psig.~~ Note 1 indicates that ~~EAL 3~~CA4.3 is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 60 minute time frame assuming that the RCS pressure ~~increase~~rise has remained less than the site specific pressure value. EAL CA4.3 does not apply when Pressurizer is solid. Minor temperature changes during solid plant operations will cause dramatic pressure swings. The described conditions, "RCS integrity not established" sets the "Pressurizer not solid" initial condition for this EAL, therefore this EAL does not apply when Pressurizer is solid.

Escalation to Site Area would be via CS1 or CS2 should boiling result in significant ~~RPV~~Reactor Vessel level loss leading to core uncovery.

~~For PWRs, t~~This IC and its associated EALs are based on 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 sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above 200 degrees~~-~~F when the heat removal function is available.

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

1. Technical Specifications Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
2. OP 1A, Cold Shutdown to Hot Standby
3. Technical Specifications B 3.6.1, Containment, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206

4. CL 1E, Containment Closure Checklist
5. OP 4F, Reactor Coolant System Reduced Inventory Requirements
6. DBD-09, Reactor Coolant System

SYSTEM MALFUNCTION

CS1

Initiating Condition -- SITE AREA EMERGENCY

Loss of ~~RPV~~Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability.

Operating Mode Applicability: Cold Shutdown

~~Example Emergency Action Levels:~~Emergency Action Levels: (CS1.1 or CS1.2)

CS1.1. With CONTAINMENT CLOSURE not established:

a. ~~RPV~~Reactor Vessel inventory as indicated by ~~RPV~~Reactor Vessel level ~~less than {site-specific level}~~0% indication on LI-447/LI-447A

OR

b. ~~RPV~~Reactor Vessel level cannot be monitored for ~~>~~GREATER THAN 30 minutes with a loss of ~~RPV~~Reactor Vessel inventory as indicated by unexplained ~~{site-specific} s~~Containment Sump A ~~and~~ or ~~[make this change for CA1/2 also]~~ Waste Holdup T~~t~~ank level ~~increase~~rise

CS1.2. With CONTAINMENT CLOSURE established:

a. ~~RPV~~Reactor Vessel inventory as indicated by ~~RPV~~Reactor Vessel level ~~less than [30 ft] 27 ft (RVLIS)TOAF~~ 0% indication on LI-447/LI-447A

OR

b. ~~RPV~~Reactor Vessel level cannot be monitored for ~~>~~GREATER THAN 30 minutes with a loss of ~~RPV~~Reactor Vessel inventory as indicated by either:
   - Unexplained ~~{site-specific} s~~Containment Sump A ~~and~~ or Waste Holdup T~~t~~ank level ~~increase~~rise
   - Erratic Source Range Monitor Indication

Basis:

Under the conditions specified by this IC, continued decrease in ~~RPV~~Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to an ~~RPV~~Reactor Vessel breach, pressure boundary leakage, or continued boiling in the ~~RPV~~Reactor Vessel.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours ~~{site-specific}~~ or longer after the reactor has been shutdown. Thus the heatup threat and therefore the

threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

In the cold shutdown mode, normal RCS level and reactor vessel level indication systems (RVLIS) will normally be available. During preparations for refueling, reactor vessel level indication may not be available. In this instance, classification would be completed under CS1.2 for sump and tank level indications. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that ~~RPV~~Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level ~~increase~~rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

These ~~example~~ EALs are based on 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.* A number of variables, ~~(BWRs – e.g., such as initial vessel level, or shutdown heat removal system design)~~ (PWRs - e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30-minutes was chosen.

The level associated with or without CONTAINMENT CLOSURE corresponds to the bottom inside diameter of the RCS loop.

~~If a PWRs RVLIS is unable to distinguish 6" below the bottom ID of the RCS loop penetration, then the first observable point below the bottom ID of the loop should be chosen as the setpoint. If a RVLIS is not available such that the PWR EAL setpoint cannot be determined, then EAL 1.b should be used to determine if the IC has been met.~~

The 30-minute duration allowed when CONTAINMENT CLOSURE is established allows sufficient time for actions to be performed to recover needed cooling equipment and is considered to be conservative given that level is being monitored via CS1 and CS2. ~~For PWRs the~~ Eeffluent release is not expected with closure established. ~~For BWRs releases would be monitored and escalation would be via Category A ICs if required.~~

Thus, ~~for both PWR and BWR~~ declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of ~~RPV~~Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the ~~RPV~~Reactor Vessel) or radiological effluent IC ~~AG1~~ RG1 (Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 ~~mR~~ mrem TEDE or 5000 ~~mR~~ mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).
~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

1. PBNP FSAR 4.0 Reactor Coolant System Design Basis
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. OI 55, Primary Leak Rate Calculation

SYSTEM MALFUNCTION

CS2

Initiating Condition -- SITE AREA EMERGENCY

Loss of ~~RPV~~Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the ~~RPV~~Reactor Vessel.

Operating Mode Applicability: Refueling

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (CS2.1 or CS2.2)

CS2.1. With CONTAINMENT CLOSURE not established:

a. ~~RPV~~Reactor Vessel inventory as indicated by ~~RPV~~Reactor Vessel level ~~less than {site-specific level}~~0% indicated on LI-447/LI-447A

OR

b. ~~RPV~~Reactor Vessel level cannot be monitored with ~~i~~Indication of core uncovery as evidenced by ~~one or more~~any of the following:
- Containment High Range Radiation Monitor reading ~~>~~GREATER THAN ~~{site-specific} setpoint~~10 R/hr
- Erratic Source Range Monitor Indication
- ~~Other {site-specific} indications~~

CS2.2. With CONTAINMENT CLOSURE established

a. ~~RPV~~Reactor Vessel inventory as indicated by ~~RPV~~Reactor Vessel level ~~less than TOAF~~0% indicated on LI-447/LI-447A

OR

b. ~~RPV~~Reactor Vessel level cannot be monitored with ~~Indication~~ indication of core uncovery as evidenced by one or more of the following:
- Containment High Range Radiation Monitor reading ~~>~~GREATER THAN ~~{site-specific} setpoint~~10 R/hr
- Erratic Source Range Monitor Indication
- ~~Other {site-specific} indications~~

Basis:

Under the conditions specified by this IC, continued decrease in ~~RPV~~Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to an ~~RPV~~Reactor Vessel breach or continued boiling in the ~~RPV~~Reactor Vessel. ~~Since BWRs have RCS penetrations below the setpoint, continued level decrease may be indicative of pressure boundary leakage.~~

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 ~~hours (site-specific)~~ or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

These ~~example~~ EALs are based on 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.* A number of variables, ~~(BWRs – e.g., such as initial vessel level, or shutdown heat removal system design) (PWRs –~~ (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. ~~Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.~~

The level associated with or without CONTAINMENT CLOSURE corresponds to the bottom inside diameter of the RCS loop. The level associated with containment closure established corresponds to the top of active fuel.

~~If a PWRs RVLIS is unable to distinguish 6" below the bottom ID of the RCS loop penetration, then the first observable point below the bottom ID of the loop should be chosen as the setpoint. If a RVLIS is not available such that the PWR EAL setpoint cannot be determined, then EAL 1.b should be used to determine if the IC has been met.~~

In Refueling mode, normal RCS level indication (e.g., RVLIS) may be unavailable but alternate means of level indication are normally ~~installed~~available to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment High Range Monitor indication and possible alarm. Typical readings at full power are 1.0 to 2.5 R/hr. The Containment High Radiation indicators (1(2)RM126, 127 and 128) are log scaled with a span of $1\text{-}10^8$ R/hr. The 10 R/hr reading has been selected to be well above that expected under normal plant conditions [Ref. 1].

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (NIS N-31 and N-32) can be used as a tool for making such determinations.

~~As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in up-scaled Containment High Range Monitor indication and possible alarm. EAL 1.b and EAL 2.b calculations should be performed to conservatively estimate a site-specific dose rate setpoint indicative of core uncovery (ie., level at TOAF). Additionally, post-TMI studies indicated that the installed 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 2~~CS2.2 in the refueling mode, normal means of ~~RPV~~Reactor Vessel level indication may not be available. Redundant means of ~~RPV~~Reactor Vessel level indication will be normally ~~installed~~ available (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

~~For PWRs the e~~Effluent release is not expected with closure established. ~~For BWRs releases would be monitored and escalation would be via Category A ICs if required.~~

Thus, ~~for both PWR and BWR~~ declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of ~~RPV~~Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the ~~RPV~~Reactor Vessel) or radiological effluent IC AG1 (Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 ~~mR~~ mrem TEDE or 5000 ~~mR~~ mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

1. Eng Eval 2001-28, Containment High Radiation Channel Check Tolerance, 10/5/01
2. PBNP FSAR 4.0 Reactor Coolant System Design Basis, Table 4.1-6

Initiating Condition -- GENERAL EMERGENCY

Loss of ~~RPV~~Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the ~~RPV~~Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

~~Example Emergency~~Emergency Action Level: ~~(1 and 2 and 3)~~

CG1.1. ~~RPV~~Reactor Vessel Level:

1. Loss of ~~RPV~~ Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Hold Up Tank level rise or any other indication of loss of Reactor Vessel inventory~~{site-specific} sump and tank level increase~~

   AND

2. ~~RPV~~ Reactor Vessel I~~L~~evel:
   a. ~~a.~~ LESS THAN [30 ft] 27 ft (RVLIS NR) or 0% indicated on LI-447/LI-447A ~~less than TOAF~~ for ~~>~~GREATER THAN -30 minutes

      OR

   b. cannot be monitored with ~~Indication~~ indication of core uncovery for ~~>~~GREATER THAN 30 minutes as evidenced by ~~one or more~~any of the following:
      - Containment High Range Radiation Monitor reading ~~>~~GREATER THAN ~~{site-specific} setpoint~~10 R/hr
      - Erratic Source Range Monitor Indication

      ~~□Other {site-specific} indications~~

      AND

3. ~~{Site-specific} indication of~~ CONTAINMENT challenged as indicated by ~~one or more~~any of the following:
   - ~~Explosive mixture inside~~GREATER THAN OR EQUAL TO 6% hydrogen concentration in containment
   - ~~Pressure~~ Containment pressure above ~~{site-specific} value~~60 psig
   - CONTAINMENT CLOSURE not established

   ~~□Secondary Containment radiation monitors above {site-specific} value (BWR only)~~

Basis:

~~For EAL 1 in the cold shutdown mode, normal RCS level and RPV level instrumentation systems will normally be available. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against~~

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

~~For EAL 1 in the refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. For both cold shutdown and refueling modes sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.~~

~~EAL 2~~This EAL represents the inability to restore and maintain ~~RPV~~Reactor Vessel level ~~to~~ above the top of active fuel. Fuel damage is probable if ~~RPV~~Reactor Vessel level cannot be restored, as available decay heat will cause boiling, further reducing the ~~RPV~~Reactor Vessel level. Setpoints enclosed in brackets (e.g., [30 ft], etc.) are used under adverse containment conditions [Ref. 3, 4].

These ~~example~~is EAL~~s~~ is~~are~~ based on 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.* A number of variables~~, (BWRs – e.g., such as initial vessel level, or shutdown heat removal system design) (PWRs –~~(e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

In the cold shutdown mode, normal RCS level and Reactor Vessel level instrumentation systems will normally be available. In the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally available (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment High Range Monitor indication and possible alarm. Typical readings at full power are 1.0 to 2.5 R/hr. The Containment High Radiation indicators (1(2)RM126, 127 and 128) are log scaled with a span of $1\text{-}10^8$ R/hr. The 10 R/hr reading has been selected to be well above that expected under normal plant conditions [Ref. 1].

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (NIS N-31 and N-32) can be used as a tool for making such determinations.

~~As water level in the RPV lowers, the dose rate above the core will increase. For most designs the dose rate due to this core shine should result in up-scaled Containment High Range Monitor indication and possible alarm. Calculations should be performed to conservatively estimate a site-specific dose rate setpoint indicative of core uncovery (ie...level at TOAF. Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (NIS N-31 and N-32) can be used as a tool for making~~

~~such determinationsAdditionally, post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.~~

The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause fuel clad failure. With the CONTAINMENT breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE.

~~In the context of EAL 3,~~ CONTAINMENT CLOSURE is the action taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. CONTAINMENT CLOSURE should not be confused with refueling containment integrity as defined in technical specifications. Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS Barrier and Fuel Clad Barrier EALs, escalation to GE would not occur.

~~The site-specific pressure at which CONTAINMENT is considered challenged may change based on the condition of the CONTAINMENT. If the Unit is in the cold shutdown mode and the CONTAINMENT is fully intact then the site-specific setpoint should be equivalent to the CONTAINMENT design pressure. This is consistent with typical owner's groups Emergency Response Procedures. If CONTAINMENT CLOSURE is established intentionally by the plant staff in preparations for inspection, maintenance, or refueling then the site-specific setpoint should be based on the site-specific pressure assumed for CONTAINMENT CLOSURE.~~

~~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 radiation monitor values should be based on the EOP "maximum safe values" because these values are easily recognizable and have an emergency basis.~~

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 mixture of dissolved gasses in CONTAINMENT. However, CONTAINMENT monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

The containment design pressure (60 psig) is well in excess of that expected from the design basis loss of coolant accident [Ref. 7].

~~Expanded basis for these assumptions is provided in Appendix C.~~

PBNP Basis Reference(s):

1. Eng Eval 2001-28, Containment High Radiation Channel Check Tolerance, 10/5/01
2. Volian Enterprises Calculation WEP-SPT-25
3. CSP-C.I UNIT 1 RED, CRITICAL SAFETY PROCEDURE SAFETY RELATED RESPONSE TO INADEQUATE CORE COOLING, Step 11
4. BG-CSP-Z.1, Response to High Containment Pressure
5. EPIP 10.3, POST-ACCIDENT CONTAINMENT HYDROGEN REDUCTION

6. FSAR Section 5.1, Containment System Structure
7. BG-CSP-ST.0, CSFST

Table E-0

Recognition Category E

Events Related to ISFSI Malfunction

INITIATING CONDITION MATRIX

~~NOUE~~

E-HU1	Damage to a loaded cask CONFINEMENT BOUNDARY. *Op. Mode: Not Applicable*
E-HU2	Confirmed security event with potential loss of level of safety of the ISFSI *Op. Mode: Not Applicable*

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EVENTS RELATED TO ISFSI

~~E-~~HEU1

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: Not applicable

~~Example Emergency~~Emergency Action Level: (EU1.1 or EU1.2 or EU1.3)

EU1.1. Any one of the following ~~N~~natural phenomena events ~~affecting~~ with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY.

Report of plant personnel of a:
- Tornado strike
- Earthquake
- Flood
- Lightning strike

~~(site-specific list)~~

EU1.2. Any of the following Accident conditions ~~affecting~~ with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY.

- Vent Blockage
- Cask Drop
- Accidental Pressurization
- Air Vent and Outlet Shielding Reduction

~~(site-specific list)~~

EU1.3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask CONFINEMENT BOUNDARY.

Basis:

A ~~N~~OUE in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

For ~~EAL #1 and EAL #~~EU1.1 and EU1.2, the results of the ISFSI Safety Analysis Report (SAR) ~~per NUREG-1536 or SAR referenced in the cask('s) Certificate of Compliance~~ and the related NRC Safety Evaluation Report ~~should~~ were ~~be~~ used to develop the site-specific list of natural phenomena events and accident conditions. These EALs ~~would~~ address responses to events defined in the ISFSI Safety Analysis Report~~a dropped cask, a tipped over cask, explosion, missile damage, fire damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.)~~.

For EAL #U1.3, any condition not explicitly detailed as an EAL threshold value, which, in the judgment of the Emergency Director, is a potential degradation in the level of safety of the ISFSI. Emergency Director judgment is to be based on known conditions and the expected response to mitigating activities within a short time period.

A UE in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. [Ref. 1, 2].

PBNP Basis Reference(s):

1. VSC-24, Conditions for Cask Use and Technical Specifications Docket No. 72-1007 Certificate of Compliance No. 1007
2. Standardized NUHOMS Horizontal Modular Storage System Certificate of Compliance No. 1004
3. Technical Specification 1.2.4 of the VSC-24 Certificate of Compliance
4. AOP-8G Ventilated Storage Cask (VSC) Drop or Tipover
5. Technical Specification 1.2.7 of the NUHOMS Certificate of Compliance
6. NUHOMS SAR
7. NUH-003 Rev 8, June 2004 "FSAR for Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel

EVENTS RELATED TO ISFSI

~~E-~~HEU2

Initiating Condition – ~~NOTIFICATION OF~~ UNUSUAL EVENT

Confirmed Security Event with potential loss of level of safety of the ISFSI.

Operating Mode Applicability: Not applicable

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

EU2.1. Security Event as determined from PBNP Physical Security Plan~~(site-specific) Security Plan~~ and reported by the ~~(site-specific) s~~Security ~~s~~Shift ~~s~~Supervisi~~on~~or.

Basis:

This EAL is based on PBNP Physical Security Plan ~~(site-specific) Security Plans.~~ Security events which do not represent a potential degradation in the level of safety of the ISFSI, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72.

Reference is made to the Security Shift Supervisor~~(site-specific) security shift supervision~~ because these individuals are the designated personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Security Plan.

PBNP Basis Reference(s):

1. PBNP Physical Security Plan

Table ~~5-F-1~~0

Recognition Category F

Fission Product Barrier Degradation

INITIATING CONDITION MATRIX

~~See Table 3 for BWR Example EALs~~
~~See Table 4 for PWR Example EALs~~

~~NOUE~~	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
FU1 ANY Loss or ANY Potential Loss of Containment *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FA1** ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FS1** Loss or Potential Loss of ANY Two Barriers *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FG1** Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*

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~~ (See Sections 3.4 and 3.8)~~. ~~NO~~UE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

   - At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

   - The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily ~~increasing~~ lowering would represent an increasing risk to public health and safety.

2. Fission Product Barrier ICs must be capable of addressing event dynamics. Thus, the EAL Reference Table ~~3 and 4~~F-1 states that imminent (i.e., within 2 hours) Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

Deleted all BWR FPB Guidance

TABLE ~~5~~-F-41

~~PWR~~-PBNP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier Example ~~EALS~~		RCS Barrier Example ~~EALS~~		Containment Barrier Example ~~EALS~~	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
Conditions requiring entry into Core Cooling ~~Red~~ RED Path (CSP-C.1)	Conditions requiring entry into Core Cooling ~~Orange~~ ORANGE Path (CSP-C.2)~~OR~~ OR Conditions requiring entry into Heat Sink ~~Red~~ RED Path (CSP-H.1)	Not Applicable	Conditions requiring entry into RCS Integrity ~~Red~~ RED Path (CSP-P.1) OR Conditions requiring entry into Heat Sink ~~Red~~ RED Path (CSP-H.1)	Not Applicable	Conditions requiring entry into Containment ~~Red~~ RED Path (CSP-Z.1)
OR		**OR**		**OR**	
2. Primary Coolant Activity Level		**2. RCS Leak Rate**		**2. Containment Pressure**	
Coolant Activity GREATER THAN 300 µCi/gm I-131 equivalent~~(site-specific)~~ ~~Value~~	Not Applicable	GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling (LESS THAN OR EQUAL TO [80 degree F] 35 degree F)	Unisolable leak exceeding ~~the capacity of one charging pump in the normal charging mode~~60 gpm	Rapid unexplained ~~decrease~~ lowering following initial ~~increase~~rise OR Containment pressure or sump level response not consistent with LOCA conditions	~~(Site-specific)~~60 PSIG and ~~increasing~~ rising OR Hydrogen concentration in containment GREATER THAN OR EQUAL TO 6%~~Explosive mixture exists~~ OR Containment pPressure GREATER ~~than containment depressurization actuation setpoint~~ THAN 25 psig with ~~less than~~LESS THAN one full train of depressurization equipment operating
OR				**OR**	
3. Core Exit Thermocouple Readings				**3. Core Exit ~~Themocouple~~Thermocouple Reading**	

TABLE ~~5~~-F-41

~~PWR~~-PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier ~~Example~~ EALS		RCS Barrier ~~Example~~ EALS		Containment Barrier ~~Example~~ EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
GREATER THAN OR EQUAL TO 1200 degree F	GREATER THAN ~~(site-specific)~~ OR EQUAL TO 700 degree F			Not applicable	Core exit thermocouples in excess of 1200 degrees F and restoration procedures not effective within 15 minutes; **OR,** core exit -thermocouples in excess of 700 degrees F with reactor vessel level below 27 ft RVLIS NR (with no RCPs running) top of active fuel and restoration procedures not effective within 15 minutes

TABLE ~~5~~-F-41

~~PWR~~-PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier ~~Example~~ EALS		RCS Barrier ~~Example~~ EALS		Containment Barrier ~~Example~~ EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
OR		**OR**		**OR**	
4. Reactor Vessel Water Level		**3. SG Tube Rupture**		**4. SG Secondary Side Release with P-to-S Leakage**	
Not Applicable	Level LESS THAN OR EQUAL TO ~~(site-specific) value~~ ▪ 25 ft RVLIS NR (with no RCPs running) • [100 ft] 90 ft RVLIS WR (with 1 RCP running) ▪ [120 ft] 110 ft RVLIS WR (with 2 RCPs running)	SGTR that results in an ECCS (SI) Actuation	Not Applicable	RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate greater than 10 gpm with nonisolable steam release from affected S/G to the environment	Not applicable
				OR	
				5. CNMT Isolation Valves Status After CNMT Isolation	
				Containment isolation valve(s) not closed ~~Valve(s) not closed~~ AND -Ddownstream pathway to the environment exists, after containment isolation	Not Applicable
OR		**OR**		**OR**	
5. Containment Radiation Monitoring		**4. Containment Radiation Monitoring**		**6. Significant Radioactive Inventory In Containment**	

TABLE ~~5~~-F-41

~~PWR~~-PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier ~~Example~~ EALS		RCS Barrier ~~Example~~ EALS		Containment Barrier ~~Example~~ EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
Containment rad monitor reading GREATER THAN ~~(site-specific)~~17 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128	Not Applicable	Containment rad monitor reading GREATER THAN 3.5 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128	Not Applicable	Not Applicable	Containment rad monitor reading GREATER THAN ~~(site-specific)~~15,900 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128

TABLE ~~5~~-F-41

~~PWR~~-PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier Example ~~EALS~~		RCS Barrier Example ~~EALS~~		Containment Barrier Example ~~EALS~~	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
OR		**OR**		**OR**	
6. Other ~~(Site-Specific)~~ Indications		5. Other ~~(Site-Specific)~~ Indications		7. Other ~~(site-specific)~~ Indications	
Failed Fuel Monitor (RE-109) reading~~(Site-specific) as applicable~~ GREATER THAN OR EQUAL TO 4500 mR~~rem~~/hr	~~(Site-specific) as applicable~~Not Applicable	~~(Site-specific) as applicable~~Not Applicable	~~(Site-specific) as applicable~~Not Applicable	~~(Site-specific) as applicable~~Not Applicable	~~(Site-specific) as applicable~~Not Applicable
OR		**OR**		**OR**	
7. Emergency Director Judgment		6. Emergency Director Judgment		8. Emergency Director Judgment	
Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier		Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier		Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	

Basis Information For Table 5-F-41
~~PWR~~ PBNP Emergency Action Level
Fission Product Barrier Reference Table

FUEL CLAD BARRIER ~~EXAMPLE~~ EALs: (1 or 2 or 3 or 4 or 5 or 6 or 7)

The Fuel Clad Barrier is the zircalloy or stainless steel tubes that contain the fuel pellets.

1. Critical Safety Function Status

~~This EAL is for PWRs using Critical Safety Function Status Tree (CSFST) monitoring and functional restoration procedures. For more information, please refer to Section 3.9 of this report.~~ RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.

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. [Ref. 1, 2]

Heat Sink - RED indicates the ultimate heat sink function is under extreme challenge and thus these two items (Core Cooling – ORANGE or Heat Sink – RED) indicate potential loss of the Fuel Clad Barrier. CSP-H.1 is the Critical Safety Procedure that provides directions to respond to a loss of secondary heat sink in both steam generators. [Ref. 1, 3]

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

CSFST setpoints enclosed in brackets (e.g., [120 ft], etc.) are used under adverse containment conditions. Adverse containment conditions are defined as:

- Containment pressure is equal to or greater than 10 psig.
- Containment radiation is currently greater than or equal to 1E5 R/hr.
- Integrated dose is greater than 1E6 R or unknown.

The barrier loss/potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

2. Primary Coolant Activity Level

This ~~(site-specific)~~ value ~~corresponds to~~ is 300 μCi/gm I_{131} equivalent. Assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost. ~~The value expressed can be either in mR/hr observed on the sample or as μCi/gm results from analysis.~~

There is no equivalent "Potential Loss" EAL for this item.

3. Core Exit Thermocouple Readings

Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked). ~~or plants which do not have a CSF scheme.~~

The "Loss" EAL 1200 degrees F~~(site-specific)~~ reading should correspond to significant superheating of the coolant. This value ~~typically~~ corresponds to the temperature reading that indicates core cooling - RED in Fuel Clad Barrier EAL #1 which is ~~usually about~~ 1200 degrees F. [Ref. 1]

The "Potential Loss" EAL 700 degrees F~~(site-specific)~~ reading should correspond to loss of subcooling. This value ~~typically~~ corresponds to the temperature reading that indicates core cooling - ORANGE in Fuel Clad Barrier EAL #1 which is ~~usually about~~ 700 ~~to 900~~ degrees F. [Ref.1]

4. Reactor Vessel Water Level

There is no "Loss" EAL corresponding to this item because it is better covered by the other Fuel Clad Barrier "Loss" EALs.

~~The (site-specific) value for the "Potential Loss" EAL corresponds to the top of the active fuel. For sites using CSFSTs, t~~The "Potential Loss" EAL is defined by the Core Cooling - ORANGE path [Ref.~~Reference~~ 1]. The 25 ft water level is a consideration in the Core Cooling - ORANGE path only after the determination is made that no RCPs are running. With one RCP running, RVLIS WR value of [100 ft] 90 ft is used. With two RCPs running RVLIS WR values of [120 ft] 110 ft are used.~~[Reference 1]The (site-specific) value in this EAL should be consistent with the CSFST value.~~

5. Containment Radiation Monitoring

The ~~(site-specific)~~17 R/hr reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading ~~should be~~is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere [Ref. 6,]. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier Loss EAL #4. Thus, this EAL indicates a loss of both the fuel clad barrier and a loss of RCS barrier.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

There is no "Potential Loss" EAL associated with this item.

6. Failed Fuel Monitor (RE-109) reading GREATER THAN 4500 mR/hr~~Other (Site-Specific) Indications~~

A Failed Fuel Monitor reading of greater than 4500 mR/hr indicates the release of reactor coolant, with elevated activity indicative of fuel damage. The reading is calculated assuming the

instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 activity into the Primary System. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage. [Ref. 8]. ~~This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the Fuel Clad barrier, including indications from containment air monitors or any other (site-specific) instrumentation.~~

There is no "Potential Loss" EAL associated with this item

7. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. ~~(See also IC-SG1, "Prolonged Loss or All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)~~

RCS BARRIER ~~EXAMPLE~~ EALs: (1 or 2 or 3 or 4 or 5 or 6)

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

~~This EAL is for PWRs using Critical Safety Function Status Tree (CSFST) monitoring and functional restoration procedures. For more information, please refer to Section 3.9 of this report.~~ RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

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. [Ref. 9, 10]

Heat Sink-Red path is entered if narrow range level in any S/G is equal to or less than [51%] 29% and total feedwater flow to S/Gs is equal to or less than 200 gpm. The combination of these two conditions indicates 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. [Ref. 11]

The barrier potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

There is no "Loss" EAL associated with this item.

2. RCS Leak Rate

The "Loss" EAL addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one ~~centrifugal~~ charging pump discharging to the charging header. A second charging pump being required is indicative of a substantial RCS leak. 60 gpm is the minimum design flow rate for each charging pump.~~For plants with low capacity charging pumps, a 50 gpm leak rate value may be used to indicate the Potential Loss.~~ [Ref. 12]

3. SG Tube Rupture

This EAL is intended to address the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier "Loss" EAL #4 and Fuel Clad Barrier EALs. The "Loss" EAL addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation of ECCS (SI). ECCS (SI) actuation is caused by:

- PZR Low Pressure (equal to or less than 1735 psig)
- Steam Line Low Pressure (equal to or less than 530 psig)
- Containment High Pressure (equal to or greater than 5 psig)

140 gpm is the design maximum capacity of all charging pumps.

This is consistent to the RCS Barrier "Potential Loss" EAL #2. ~~For plants that have implemented W.O.G. emergency response guides, t~~This condition is described by "entry into EOP-3 required by EOPs". By itself, this EAL will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment Barrier "Loss" EAL #4. [Ref. 14, 15]

There is no "Potential Loss" EAL.

4. Containment Radiation Monitoring

The ~~(site-specific)~~3.5 R/hr reading is a value which indicates the release of reactor coolant to the containment. The reading ~~should be~~is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within ~~T/S~~Technical Specifications) into the containment atmosphere~~.~~ [Ref. 6]

This reading ~~will be~~is less than that specified for Fuel Clad Barrier EAL #5. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL #5, fuel damage would also be indicated.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

~~However, if the site-specific physical location of the containment radiation monitor is such that radiation from a cloud of released RCS gases could not be distinguished from radiation from nearby piping and components containing elevated reactor coolant activity, this EAL should be omitted and other site specific indications of RCS leakage substituted.~~

There is no "Potential Loss" EAL associated with this item.

5. Other (Site-Specific) Indications

~~This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the RCS barrier, including indications from containment air monitors or any other (site-specific) instrumentation.~~None

6. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. ~~(See also IC-SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)~~

CONTAINMENT BARRIER ~~EXAMPLE~~ EALs: (1 or 2 or 3 or 4 or 5 or 6 or 7 or 8)

The Containment Barrier includes the containment building, its 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.

1. Critical Safety Function Status

~~This EAL is for PWRs using Critical Safety Function Status Tree (CSFST) monitoring and functional restoration procedures. For more information, please refer to Section 3.9 of this report.~~ RED path indicates an extreme challenge to the safety function. Containment-Red path is entered if containment pressure is equal to or greater than 60 psig. This pressure is the containment design pressure and is well in excess of that expected from the design basis loss of coolant accident~~derived from appropriate instrument readings and/or sampling results~~, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier. CSP-Z.1 is the Critical Safety Procedure that provides directions to respond to high containment pressure. [Ref. 16, 17, 18]

The barrier potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

There is no "Loss" EAL associated with this item.

2. Containment Pressure

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure ~~increase~~rise indicates a loss of containment integrity. FSAR Figure 14.3.2-1 illustrates containment pressure response for a bounding LOCA. Containment pressure peaks at approximately 34 psia. [Ref. 25, 26, 27, 28]

Containment pressure ~~and~~or sump levels should ~~increase~~rise as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure not ~~increasing~~-rising indicates containment bypass and a loss of containment integrity.

The ~~(site-specific)~~60 PSIG for potential loss of containment is based on the containment design pressure. [Ref. 24, 25, 26, 27]

If hydrogen concentration reaches or exceeds 6% in an oxygen rich environment, an explosive mixture exists. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred~~Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to the RED path in EAL #1 above and may be declared by those sites using CSFSTs.~~ As described above, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier. [Ref. 21, 22]

The second potential loss EAL represents a potential loss of containment in that the containment heat removal/depressurization system~~(e.g., containment sprays, ice condenser fans, etc.,~~ (but not

including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint (25 psig) at which the equipment was supposed to have actuated. 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. [Ref. 16, 20, 23]

3. Core Exit Thermocouples

In this EAL, the ~~function~~ restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is ~~decreasing~~ lowering or if the vessel water level is ~~increasing~~rising. ~~For units using the CSF status trees a direct correlation to those status trees can be made if the effectiveness of the restoration procedures is also evaluated as stated below.~~

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective. ~~The reactor vessel level chosen should be consistent with the emergency response guides applicable to the facility.~~

The conditions in this potential loss EAL represent an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the Core Cooling and Heat Sink criteria in the Fuel and RCS barrier columns, this EAL would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path. [Ref. 1, 5]

There is no "Loss" EAL associated with this item.

4. SG Secondary Side Release With Primary To Secondary Leakage

This "loss" EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The first "loss" EAL addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers. In conjunction with RCS Barrier "loss" EAL #3, this would always result in the declaration of a Site Area Emergency. A faulted S/G means the existence of secondary side leakage that results in an uncontrolled lowering in steam generator pressure or the steam generator being completely depressurized. A ruptured S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection. Confirmation should be based on diagnostic activities consistent with EOP-0, Reactor Trip or Safety Injection.

The second "loss" EAL addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for

establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs. [Ref. 14]

~~Users~~ It should be realized that the two "loss" EALs described above could be considered redundant. This was recognized during the development process. The inclusion of an EAL that uses ~~Emegency~~Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

~~The leakage threshold for this EAL has been increased with Revision 3. In the earlier revision, the threshold was leakage greater than T/S allowable. Since the prior revision, many plants have implemented reduced steam generator T/S limits (e.g., 150 gpd) as a defense in depth associated with alternate steam generator plugging criteria. The 150 gpd threshold is deemed too low for use as an emergency threshold.~~ A pressure boundary leakage of 10 gpm ~~was~~ is used as the threshold in IC SU5.1, RCS Leakage, and is deemed appropriate for this EAL. For smaller breaks, not exceeding the normal charging capacity threshold in RCS Barrier "Potential Loss" EAL #2 (RCS Leak Rate) or not resulting in ECCS actuation in EAL #3 (SG Tube Rupture), this EAL results in a ~~NO~~UE. For larger breaks, RCS barrier EALs #2 and #3 would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this EAL would exist in conjunction with RCS barrier "Loss" EAL #3 and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

There is no "Potential Loss" EAL associated with this item.

5. Containment Isolation Valve Status After Containment Isolation

This EAL is intended to address incomplete containment isolation that allows direct release to the environment. It represents a loss of the containment barrier.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no "Potential Loss" EAL associated with this item.

6. Significant Radioactive Inventory in Containment

The ~~(site-specific)~~15,900 R/hr reading is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS Barriers. [Ref. 6] ~~As stated in Section 3.8, a~~A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

Monitors used for this fission product barrier potential loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

~~Unless there is a (site-specific) analysis justifying a higher value, it is recommended that a radiation monitor reading corresponding to 20% fuel clad damage be specified here.~~

There is no "Loss" EAL associated with this item.

7. Other (Site-Specific) Indications

~~This EAL should cover other (site-specific) indications that may unambiguously indicate loss or potential loss of the containment barrier, including indications from area or ventilation monitors in containment annulus or other contiguous buildings. If site emergency operating procedures provide for venting of the containment during an emergency as a means of preventing catastrophic failure, a Loss EAL should be included for the containment barrier. This EAL should be declared as soon as such venting is imminent. Containment venting as part of recovery actions is classified in accordance with the radiological effluent ICs.~~None

8. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. ~~(See also IC-SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)~~

PBNP Basis Reference(s):

1. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
2. CSP-C.2, Response to Degraded Core Cooling
3. CSP-H.1, Response to Loss of Secondary Heat Sink
4. Volian Enterprises Calculation No. WEP-SPT-25, Reactor Vessel Level EOP Setpoints
5. CSP-C.1, Response to Inadequate Core Cooling
6. Calculation 2004-0006, Dose Rate Calculation for Containment High Range and Refueling Floor Area Radiation Monitors Under Accident Conditions
7. SAMG SAG-5, Reduce Fission Product Releases, Attachment D
8. Calc 2004-0008, Failed Fuel Monitor (RE-109) Reading / Fuel Damage Correlation
9. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 4
10. CSP-P.1, Response to Imminent Pressurized Thermal Shock Condition
11. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 3, Heat Sink
12. DBD-04, Chemical and Volume Control System, Section 3.9

13. BG-CSP-ST.0 Step ST-2, Critical Safety Status Trees
14. EOP-0, Reactor Trip Or Safety Injection
15. DBD-04, Chemical and Volume Control System
16. CSP-ST.0, Unit 1(2) Critical Safety Function Status Trees, Figure 5
17. BG-CSP-ST.0 Step ST-5
18. CSP-Z.1, Response to High Containment Pressure
19. FSAR Section 5.1, Containment System Structure
20. BG-CSP-ST.0, CSFST, Step F.0.5
21. CSP-C.1 Unit 1 Red, Critical Safety Procedure Safety Related Response To Inadequate Core Cooling, Step 11
22. EPIP 10.3, Post-Accident Containment Hydrogen Reduction
23. TS B 3.6.6, Containment Spray and Cooling Systems, pgs B 3.6.6-4 & -5, 10/20/02
24. FSAR Figure 14.3.2-1, Containment Pressure Curve used in PBNP BELOCA Analysis
25. FSAR Tables 14.3.2-1 through 14.3.2-3, Large Break Loss of Coolant Accident Analysis Data
26. FSAR 14.3.1, Small Break Loss of Coolant Accident Analysis
27. FSAR 14.3.2, Large Break Loss-Of-Coolant Accident Analysis
28. CSP-Z.1, Attachment B, Containment Isolation Valves
29. CALC WEP-SPT-12, EOP S1 Reduction Analysis 6/25/99
30. STPT 2.1, Safety Injection, Rev 2., dated 10/15/96

TABLE 5-H-10

Recognition Category H

Hazards and Other Conditions Affecting Plant Safety

INITIATING CONDITION MATRIX

NOUE		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
HU1	Natural and Destructive Phenomena Affecting the PROTECTED AREA. *Op. Modes: All*	HA1	Natural and Destructive Phenomena Affecting the Plant VITAL AREA. *Op. Modes: All*				
HU2	FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection. *Op. Modes: All*	HA2	FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown. *Op. Modes: All*				
HU3	Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant. *Op. Modes: All*	HA3	Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Safety Systems Required to Establish or Maintain Safe Shutdown. *Op. Modes: All*				
HU4	Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant. *Op. Modes: All*	HA4	Confirmed Security Event in a Plant PROTECTED AREA. *Op. Modes: All*	HS1	Confirmed Security Event in a Plant VITAL AREA. *Op. Modes: All*	HG1	Security Event Resulting in Loss Of Physical Control of the Facility. *Op. Modes: All*
HU5	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a ~~NOUE~~UE. *Op. Modes: All*	HA6	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. *Op. Modes: All*	HS3	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency. *Op. Modes: All*	HG2	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency. *Op. Modes: All*
		HA5	Control Room Evacuation Has Been Initiated. *Op. Modes: All*	HS2	Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established. *Op. Modes: All*		

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Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level: (HU1.1 or HU1.2 or HU1.3 or HU1.4 or HU1.5 or HU1.6 or HU1.7)

HU1.1. Earthquake felt in plant as indicated by: ~~[(Site-Specific) method of indicatinges felt earthquake.]~~

Activation of 2 or more seismic monitors

AND

Verified by:

- Actual ground shaking

  OR

- By contacting the U.S. Geological Survey National Earthquake Information Center

HU1.2. Report by plant personnel of tornado or high winds ~~greater than~~GREATER THAN [108~~(site-specific, FSAR design basis])~~ mph (15 minute average) striking within PROTECTED AREA boundary.

HU1.3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary.

HU1.4. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

HU1.5. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

HU1.6. Uncontrolled flooding in the following~~(site-specific)~~ areas of the plant that has the potential to affect safety related equipment needed for the current operating mode:

- Auxiliary building caused by rupture of the SW header

  OR

- Water intake structure caused by rupture of a circulating water system expansion joint or fire water main. ~~[site-specific list of areas].~~

HU1.7. ~~(Site-Specific) occurrences affecting the PROTECTED AREA.~~Lake (forebay) level GREATER THAN OR EQUAL TO 8.0 ft (587.2 ft IGLD)

Basis:

~~NO~~UE in this IC are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the EALs define the location of the event based on the potential for damage to equipment contained therein. Escalation of the event to an Alert occurs when the magnitude of the event is sufficient to result in damage to equipment contained in the specified location.

~~EAL #1~~HU1.1~~ should be developed on site-specific basis~~. Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. Method of detection ~~can be~~is based on instrumentation, validated by a reliable source (U.S. Geological Survey National Earthquake Information Center), or operator assessment [Ref. 1, 2, 3, 4]. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a *"felt earthquake"* is:

> An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01g.

~~EAL #2~~HU1.2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind site specific value ~~in EAL#2 should be~~is based on ~~site-specific~~the FSAR design basis sustained wind speed of 108 mph. Wind loads of this magnitude can cause damage to safety functions. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert. [Ref. 5, 6]

~~EAL #3~~HU1.3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. If the crash is confirmed to affect a plant VITAL AREA, the event may be escalated to Alert. [Ref. 6]

For ~~EAL #4~~HU1.4 only those EXPLOSIONs of sufficient force to damage permanent structures or equipment within the PROTECTED AREA should be considered. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The Emergency director also needs to consider any security aspects of the EXPLOSION, if applicable. [Ref. 6]

~~EAL #5~~HU1.5 is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIREs and flammable gas build up are appropriately classified via HU2 and HU3. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant. This EAL is consistent with the definition of a ~~NOUE~~UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency

classification is based on potential damage done by missiles generated by the failure or ~~by the radiological releases for a BWR, or~~ in conjunction with a steam generator tube rupture~~, for a PWR~~. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs. [Ref. 8, 9]

~~EAL #6~~HU1.6 addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The auxiliary building and water intake structure are the vulnerable areas indicated in the IPE that contain~~The site-specific areas include those areas that contain~~ systems required for safe shutdown of the plant, that are not designed to be wetted or submerged. Escalation of the emergency classification is based on the damage caused or by access restrictions that prevent necessary plant operations or systems monitoring. [Ref. 11, 12]~~The plant's IPEEE may provide insight into areas to be considered when developing this EAL.~~

~~EAL #7~~ HU1.7 covers other site-specific phenomena such as ~~hurricane,~~ flood, or seiche. This~~ese~~ EAL~~s~~ can also be precursors of more serious events. Lake water level GREATER THAN OR EQUAL TO 8 feet (587.2 ft elevation IGLD) corresponds to the Turbine Building floor elevation. Both the Turbine Building and Circ Water Pumphouse are specifically susceptible to external flooding.~~In particular, sites subject to severe weather as defined in the NUMARC station blackout initiatives, should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.).~~

PBNP Basis Reference(s):

1. AOP-28 Seismic Event
2. FSAR Section 2.9 Seismology
3. STPT 22.1 Seismic Event Monitoring
4. EPRI document, "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989
5. FSAR Section 2.6, Meteorology
6. Bechtel Drawing C-3 Plant Areas
7. ARP 1(2) C33 1-2 Hydrogen Pressure High-Low Alarm
8. ARP 1(2) C33 1-3 Hydrogen Supply Pressure Low Alarm
9. AOP-5A Loss of Condenser Vacuum
10. NPC95-00559, PBNP Individual Plant Examination (IPE) for internal events and internal flood.
11. DG-C02, Internal Flooding
12. DBD-T41, Module A, "Hazards-Internal and External Flooding"
13. FSAR Section 2.5, Hydrology
14. International Great Lakes Datum (IGLD)

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level:

HU2.1. FIRE in- buildings or areas contiguous to any of the ~~the~~ Table H-1~~following (site-specific)~~ areas not extinguished within 15 minutes of control room notification or verification of a control room alarm~~.:~~

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

~~(Site-specific) list~~

Basis:

The purpose of this IC is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, Detection is visual observation and report by plant personnel or sensor alarm indication. The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. Verification of a fire detection system alarm includes actions that can be taken with the control room or other nearby site-specific location to ensure that the alarm is not spurious. A verified alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket). ~~The site-specific list should be limited and applies to buildings and areas contiguous (in actual contact with or immediately adjacent) to plant VITAL AREAs or other significant buildings or areas. The intent of this IC is not to include buildings (i.e., warehouses) or areas that are not contiguous (in actual contact with or immediately adjacent) to plant VITAL AREAs.~~ This EAL excludes ~~FIREs within administration buildings,~~ waste-basket FIREs~~,~~ and other small FIREs of no safety consequence.

Escalation to a higher emergency class is by IC HA4, "FIRE Affecting the Operability of Plant Safety Systems Required for the Current Operating Mode".

PBNP Basis Reference(s):

1. Bechtel Drawing C-3 Plant Areas

Initiating Condition ~~– NOTIFICATION OF~~ UNUSUAL EVENT

Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.

Operating Mode Applicability: All

~~Example Emergency Action Levels:~~Emergency Action Levels: (HU3.1 or HU3.2)

HU3.1. Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.

HU3.2. Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

This IC is based on the existence of uncontrolled releases of toxic or flammable gas that may enter the site boundary and affect normal plant operations. It is intended that releases of toxic or flammable gases are of sufficient quantity, and the release point of such gases is such that normal plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. The EALs are intended to not require significant assessment or quantification. The IC assumes an uncontrolled process that has the potential to affect plant operations, or personnel safety.

Escalation of this EAL is via HA3, which involves a quantified release of toxic or flammable gas affecting VITAL AREAs.

PBNP Basis Reference(s):

None

Initiating Condition ~~– NOTIFICATION OF~~ UNUSUAL EVENT

Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant.

Operating Mode Applicability: All

~~Example Emergency Action Levels:~~Emergency Action Levels: (HU4.1 or HU4.2)

~~1. Security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision~~

HU4.1. ~~2.~~ Security ~~[Supervision title]~~Shift Supervisor reports ANY of the following:~~A credible site-specific security threat notification.~~

- Suspected sabotage device discovered within the plant Protected Area
- Suspected sabotage device discovered outside the Protected Area or in the plant switchyard
- Confirmed tampering with safety-related equipment
- A hostage situation that disrupts NORMAL PLANT OPERATIONS
- Civil disturbance or strike which disrupts NORMAL PLANT OPERATIONS
- Internal disturbance that is <u>not</u> a short lived or that is not a harmless outburst involving ANY individuals within the Protected Area
- Malevolent use of a vehicle outside the Protected Area which disrupts NORMAL PLANT OPERATIONS

HU4.2 Notification of a credible site-specific threat by the Security Shift Supervisor or outside agency (e.g., NRC, military or law enforcement)

Basis:

Reference is made to ~~(site-specific) security shift supervision~~the Security Shift Supervisor because ~~these~~ this individuals ~~are~~ is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security~~Safeguards Contingency~~ Plan.

~~This EAL 1~~ HU4.1 is based on ~~(site-specific) Site Security Plans~~Physical Security Plan. Security events which do not represent a potential degradation in the level of safety of the plant, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Examples of security events that indicate Potential Degradation in the Level of Safety of the Plant are provided ~~below~~ for consideration.

~~Consideration should be given to the following types of events when evaluating an event against the criteria of the site specific Security Contingency Plan: SABOTAGE, HOSTAGE / EXTORTION, CIVIL DISTURBANCE, and STRIKE ACTION.~~

INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE would result in EAL escalation to an ALERT.

The intent of ~~EAL 2~~ HU4.2 is to ensure that appropriate notifications for the security threat are made in a timely manner. Only the plant to which the specific threat is made need declare the ~~Notification of an~~ Unusual Event.

The determination of "credible" is made through use of information found in the ~~(site-specific)~~ Physical Security Plan~~Safeguards Contingency Plan~~.

A higher initial classification could be made based upon the nature and timing of the threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification in accordance with the ~~[site-security-specific]~~ Physical Security Plan~~Safeguards Contingency Plan~~ and Emergency Plans.

PBNP Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. PBNP Physical Security Plan
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. NMC fleet Security Threat Assessment Policy

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition ~~NOTIFICATION OF~~ UNUSUAL EVENT

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a ~~NOUE~~UE.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level:

HU5.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Basis:

This EAL is intended to address 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 ~~NOUE~~UE emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency response procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

PBNP Basis Reference(s):

None

Initiating Condition -- ALERT

Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability: All

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (HA1.1 or HA1.2 or HA1.3 or HA1.4 or HA1.5 or HA1.6)

HA1.1. Seismic event GREATER THAN Operating Basis Earthquake (OBE) as indicated by:~~(Site-Specific) [method(s) of indicatinges Seismic Event greater than Operating Basis Earthquake (OBE).]~~

VALID seismic monitor- indication of ground acceleration **EITHER**:

GREATER THAN OR EQUAL TO 0.06 g horizontal

OR

GREATER THAN OR EQUAL TO 0.04 g vertical

HA1.2. Tornado or high winds (15 minute average) GREATER THAN~~greater than [(site-specific, FSAR design basis]~~108~~)~~ mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures / equipment or Control Room indication of degraded performance of those systems.

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

~~☐Reactor Building~~
~~☐Intake Building~~
~~☐Ultimate Heat Sink~~
~~☐Refueling Water Storage Tank~~
~~☐Diesel Generator Building~~
~~☐Turbine Building~~
~~☐Condensate Storage Tank~~
~~☐Control Room~~
~~☐Other (Site-Specific) Structures.~~

HA1.3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or control room indication of degraded performance of those systems:

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

~~☐Reactor Building~~
~~☐Intake Building~~
~~☐Ultimate Heat Sink~~
~~☐Refueling Water Storage Tank~~
~~☐Diesel Generator Building~~
~~☐Turbine Building~~
~~☐Condensate Storage Tank~~
~~☐Control Room~~
~~☐Other (Site-Specific) Structures.~~

HA1.4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas:

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

~~(site-specific) list.~~

HA1.5. Uncontrolled flooding in ~~(site-specific)~~following areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

-Auxiliary building caused by rupture of the SW header

OR

Water intake structure caused by rupture of a circulating water system expansion joint or fire water main.

HA1.6 Lake (forebay) level GREATER THAN OR EQUAL TO 9.0 ft (588.2 ft IGLD)

Basis:

The EALs in this IC escalate from the ~~NOUE~~UE EALs in HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to

classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other ICs (e.g., System Malfunction).

~~EAL #1~~HA1.1 is based on the FSAR operating basis earthquake (OBE) of 0.06 g horizontal or 0.04 g vertical acceleration. ~~should be based on site-specific FSAR design basis.~~ Seismic events of this magnitude can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. [[Ref. 1, 2, 3, 4]~~See EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, for information on seismic event categories.~~

~~EAL #2~~HA1.2 is based on the FSAR design basis sustained wind speed of 108 mph~~should be based on site-specific FSAR design basis~~. Wind loads of this magnitude can cause damage to safety functions.

Wind speed is measured as the 15 minute average wind speed. This EAL addresses events that may have resulted in a plant VITAL AREA being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant structures, systems or equipment. [Ref. 5, 6] The list of plant structures/equipment contains functions required for safe shutdown of the plant. Ultimate Heat Sink is not included in this list as there are no events identified which would adversely impact Lake Level and the system pumps and associated equipment which are located in the Intake Building and other listed structures.

~~EAL #s 2, 3, 4, 5 should specify site-specific structures or areas containing systems and functions required for safe shutdown of the plant.~~

~~EAL #3~~HA1.3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. The list of plant structures/equipment contains functions required for safe shutdown of the plant. Ultimate Heat Sink is not included in this list as there are -no events identified which would adversely impact Lake Level and the system pumps and associated equipment, which are located in the Intake Building and other listed structures.

~~EAL #~~HA1.4 is intended to address the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. ~~This site-specific list of areas should include all areas containing safety-related equipment, their controls, and their power supplies.~~ This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant. [Ref. 7, 8, 9] The list of plant structures/equipment containing functions required for safe shutdown of the plant. Ultimate Heat Sink is not included in this list as there are no events identified which would adversely impact Lake Level and the system pumps associated equipment, which are located in the Intake Building and other listed structures.

~~EAL #5~~HA1.5 addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The auxiliary building and water intake structure are those areas identified in the IPE that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged ~~The site-specific areas includes those~~

~~areas that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged~~. [Ref. 10]
~~The plant's IPEEE may provide insight into areas to be considered when developing this EAL.~~

~~EAL #6~~ HA1.6 covers high lake (forebay) 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 GREATER THAN OR EQUAL TO 9.0 feet (588.2 ft elevation IGLD) corresponds to that which can result in flooding of vital areas containing safe shutdown equipment. ~~other site-specific phenomena such as hurricane, flood, or seiche. These EALs can also be precursors of more serious events.~~

PBNP Basis Reference(s):

1. AOP-28 Seismic Event
2. FSAR Section 2.9 Seismology
3. STPT 22.1 Seismic Event Monitoring
4. EPRI document, "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989
5. FSAR Section 2.6, Meteorology
6. Bechtel Drawing C-3 Plant Areas
7. ARP 1(2) C33 1-2 Hydrogen Pressure High-Low Alarm
8. ARP 1(2) C33 1-3 Hydrogen Supply Pressure Low Alarm
9. AOP-5A Loss of Condenser Vacuum
10. NPC95-005559 PBNP Individual Plant Examination (IPE) for internal events and internal flood.
11. FSAR Section 2.5, Hydrology
12. International Great Lakes Datum (IGLD)

Initiating Condition -- ALERT

FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level:

HA2.1. FIRE or EXPLOSION in any of the following ~~(site-specific)~~ areas (Table H-1):

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

~~(Site-specific) list~~

AND

Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.

Basis:

These areas contain systems and components required for the safe shutdown functions of the plant. The PBNP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode. ~~Site-specific areas containing functions and systems required for the safe shutdown of the plant should be specified. Site-Specific Safe Shutdown Analysis should be consulted for equipment and plant areas required to establish or maintain safe shutdown.~~ This will make it easier to determine if the FIRE or EXPLOSION is potentially affecting one or more redundant trains of safety systems. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs.

This EAL addresses a FIRE / EXPLOSION and not the degradation in performance of affected systems. System degradation is addressed in the System Malfunction EALs. The reference to damage of systems is used to identify the magnitude of the FIRE / EXPLOSION and to discriminate against minor FIREs / EXPLOSIONs. The reference to safety systems is included to discriminate against FIREs / EXPLOSIONs in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the

FIRE / EXPLOSION was large enough to cause damage to these systems. ~~Thus, the designation of a single train was intentional and is appropriate when the FIRE / EXPLOSION is large enough to affect more than one component.~~

This situation is not the same as removing equipment for maintenance that is covered by a plant's Technical Specifications. Removal of equipment for maintenance is a planned activity controlled in accordance with procedures and, as such, does not constitute a substantial degradation in the level of safety of the plant. A FIRE / EXPLOSION is an UNPLANNED activity and, as such, does constitute a substantial degradation in the level of safety of the plant. In this situation, an Alert classification is warranted.

The inclusion of a "report of VISIBLE DAMAGE" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform these damage assessments. The Emergency Director also needs to consider any security aspects of the EXPLOSIONs, if applicable.

PBNP Basis Reference(s):

None

Initiating Condition -- ALERT

Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (HA3.1 or HA3.2)

HA3.1. Report or detection of toxic gases within or contiguous to a ~~Safe Shutdown~~/VITAL AREA (Table H-~~21)~~~~VITAL AREA~~ in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

HA3.2. Report or detection of gases in concentration GREATER THAN the LOWER FLAMMABILITY LIMIT within or contiguous to a ~~Safe Shutdown~~/VITAL AREA (Table H-1)~~VITAL AREA~~.

Table H-1 VITAL AREA
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

Basis:

This IC is based on gases that affect the safe operation of the plant. This IC applies to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas ~~(i.e., service water pump house)~~. The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to plant VITAL AREAs. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radioactive Effluent, or Emergency Director Judgment ICs.

~~EAL #1~~HA3.1 is met if measurement of toxic gas concentration results in an atmosphere that is IDLH within a VITAL AREA or any area or building contiguous to VITAL AREA. Exposure to an IDLH atmosphere will result in immediate harm to unprotected personnel, and would preclude access to any such affected areas.

~~EAL #2~~HA3.2 is met when the flammable gas concentration in a VITAL AREA or any building or area contiguous to a VITAL AREA exceed the LOWER FLAMMABILITY LIMIT. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL addresses concentrations at which gases can ignite/support combustion. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Once it has been determined that an uncontrolled release is occurring, then sampling must be done to determine if the concentration of the released gas is within this range.

PBNP Basis Reference(s):

None

Initiating Condition – ALERT

Confirmed Security Event in a Plant PROTECTED AREA.

Operating Mode Applicability: All

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (HA4.1 or HA4.2)

HA4.1. INTRUSION into the ~~plant~~ PROTECTED AREA by a HOSTILE FORCE.

HA4.2. Security ~~[Supervision title]~~Shift Supervisor reports any of the following:

- Sabotage device discovered in the PROTECTED AREA
- Standoff attack on the site protected area by a HOSTILE FORCE (i.e., Sniper)
- ANY Security event of increasing severity that persists for >GREATER THAN 30 minutes:
  - Credible bomb threats
  - Extortion
  - Suspicious Fire or Explosion
  - Significant Security System Hardware Failure
  - Loss of Guard Post Contact

~~Other security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision~~

Basis:

This class of security events represents an escalated threat to plant safety above that contained in the ~~NOUE~~UE. A confirmed INTRUSION report is satisfied if physical evidence indicates the presence of a HOSTILE FORCE within the PROTECTED AREA.

The Physical Security Plan identifies numerous events/conditions that constitute a threat/compromise to station security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered.~~Consideration should be given to the following types of events when evaluating an event against the criteria of the site specific Security Contingency Plan: SABOTAGE, HOSTAGE / EXTORTION, and STRIKE ACTION. The Safeguards Contingency Plan identifies numerous events/conditions that constitute a threat/compromise to a Station's security. Only those events that involve Actual or Potential Substantial degradation to the level of safety of the plant need to be considered. The following events would not normally meet this requirement; (e.g., Failure by a Member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).~~

INTRUSION into a VITAL AREA by a HOSTILE FORCE will escalate this event to a Site Area Emergency.

Reference is made to ~~(site-specific)~~ ~~s~~Security ~~s~~Shift ~~s~~Supervisi~~on~~or because ~~these~~ this individuals ~~are~~ is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy

HA5

Initiating Condition -- ALERT

Control Room Evacuation Has Been Initiated.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level:

HA5.1. Entry into AOP-10 Control Room Inaccessibility[~~(site-specific) procedure number(s) and title(s)~~] for control room evacuation.

Basis:

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facility is necessary. The AOP-10 series of procedures provide specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations. Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

PBNP Basis Reference(s):

1. AOP-10 Control Room Inaccessibility
2. AOP-10A Safe Shutdown - Local Control
3. AOP-10B U1(2) Safe to Cold Shutdown in Local Control

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

Initiating Condition -- ALERT

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level:

HA6.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels-.

Basis:

This EAL is intended to address 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 Alert emergency class.

PBNP Basis Reference(s):

None

Initiating Condition – SITE AREA EMERGENCY

Confirmed Security Event in a Plant VITAL AREA.

Operating Mode Applicability: All

~~**Example Emergency Action Levels:**~~**Emergency Action Levels:** (HS1.1 or HS1.2)

HS1.1. INTRUSION into the plant VITAL AREA by a HOSTILE FORCE.

HS1.2. Security Supervision reports confirmed sabotage discovered in a VITAL AREA ~~Other security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision~~

Basis:

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC in that a HOSTILE FORCE has progressed from the PROTECTED AREA to a VITAL AREA.

Consideration ~~should be~~is given to the following types of events when evaluating an event against the criteria of the site specific ~~Security Contingency~~Physical Security Plan: SABOTAGE and HOSTAGE / EXTORTION. The Physical Security~~Safeguards Contingency~~ Plan identifies numerous events/conditions that constitute a threat/compromise to a Station's security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered. ~~The following events would not normally meet this requirement; (e.g., Failure by a Member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).~~

Loss of Plant Control would escalate this event to a GENERAL EMERGENCY.

Reference is made to ~~(site-specific) s~~ Security ~~s~~Shift ~~s~~Supervision~~or~~ because ~~these~~ this individuals ~~are~~ is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan.

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy

Initiating Condition – SITE AREA EMERGENCY

Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level:

HS2.1. Control room evacuation has been initiated.

AND

Control of the plant cannot be established per AOP-10A Safe Shutdown - Local Control[~~(site-specific) procedure number(s) and name(s)~~] within [~~(site-specific~~]15~~)~~ minutes.

Basis:

Expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. ~~Site-specific time for transfer based on analysis or assessments as to how quickly control must be reestablished without core uncovering and/or core damage. This time should not exceed 15 minutes without additional justification.~~ The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the ~~site-specific~~ time for transfer that the ~~licensee~~ operator has control of the plant from the remote shutdown panel.

The AOP-10 series of procedures provide specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory~~reactor water level~~ (ability to cool the core), and secondary heat removal~~decay heat removal~~ (ability to maintain a heat sink)~~ for a BWR~~. ~~The equivalent functions for a PWR are reactivity control, RCS inventory, and secondary heat removal.~~

Escalation of this event, if appropriate, would be by Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Emergency Director Judgment ICs.

PBNP Basis Reference(s):

1. AOP-10 Control Room Inaccessibility
2. AOP-10A Safe Shutdown - Local Control
3. AOP-10B Safe to Cold Shutdown in Local Control
4. BG AOP-10A, Safe Shutdown – Local Control

HS3

Initiating Condition – SITE AREA EMERGENCY

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

HS3.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

This EAL is intended to address 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 class description for Site Area Emergency.

PBNP Basis Reference(s):

None

Initiating Condition – GENERAL EMERGENCY

Security Event Resulting in Loss Of Physical Control of the Facility.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level:

HG1.1. A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of **EITHER**:

A VITAL AREA such that operation of equipment required for safe shutdown is lost

OR

Spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in the pool).

Basis:

This IC encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAs (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location. Typically, these safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) ~~reactor water level~~RCS inventory (ability to cool the core), and ~~decay~~ secondary heat removal (ability to maintain a heat sink)~~ for a BWR. The equivalent functions for a PWR are reactivity control, RCS inventory, and secondary heat removal.~~ If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the above initiating condition is not met.

This EAL ~~should also~~ addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in pool).

Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se.
~~Design of the remote shutdown capability and the location of the transfer switches should be taken into account.~~

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy

HG2

Initiating Condition – GENERAL EMERGENCY

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level:

HG2.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL is intended to address 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 General Emergency class.

PBNP Basis Reference(s):

None

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Table S-0

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

~~NOUE~~		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
SU1	Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA5**	AC ~~power~~ Power ~~capability~~ Capability to ~~essential~~ Essential ~~busses~~ Busses Reduced to a Single Power Source for GREATER THAN 15 minutes Such That Any Additional Single Failure Would Result in Station Blackout. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS1**	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SG1**	Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*
		SA2	Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor ~~Scram~~Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual ~~Scram~~Trip Was Successful. *Op. Modes: Power Operation, Startup, Hot Standby*	**SS2**	Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor ~~Scram~~Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual ~~Scram~~Trip Was NOT Successful. *Op. Modes: Power Operation, Startup*	**SG2**	Failure of the Reactor Protection System to Complete an Automatic ~~Scram~~Trip and Manual ~~Scram~~Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core. *Op. Modes: Power Operation, Startup*
SU2	Inability to Reach Required Shutdown Within Technical Specification Limits. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA3**	*Deleted*	**SS4**	Complete Loss of Heat Removal Capability. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*		
SU3	UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN 15 Minutes *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA4**	UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS6**	Inability to Monitor a SIGNIFICANT TRANSIENT in Progress. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*		

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

SU7 *Deleted*

SU4 Fuel Clad Degradation.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU5 RCS Leakage.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU8 Inadvertent Criticality.
Op Modes: Hot Standby, Hot Shutdown

SA1 *Deleted*

SS3 Loss of All Vital DC Power.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SS5 *Deleted*

SYSTEM MALFUNCTION

SU1

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses for GREATER THAN~~reater Than~~ 15 Minutes.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SU1.1. Loss of all **offsite** power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 ~~(site-specific) transformers~~ for GREATER THAN~~greater than~~ 15 minutes.

AND

At least 1~~(site-specific)~~ emergency generators ~~are~~is supplying power to an emergency bus.~~ses.~~

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The 4160 VAC system includes two safety-related (essential) buses per unit, 1(2)-A05 (A train) and 1(2)-A06 (B train). Offsite power from the 345 KVAC system is stepped down through the 13.8 KVAC system to the Low Voltage Station Auxiliary Transformer (LVSAT) 1(2)-X04. The LVSATs provide power to 4160 VAC switching buses 1(2)-A03 and 1(2)-A04, which in turn provide power to safety-related buses 1(2)-A05 and 1(2)-A06.

During emergency or abnormal situations, the 4160 VAC system is supplied by emergency diesel generators (G01 through G04) or the gas turbine generator (G05). Following a loss of power, ECA 0.0 provides guidance to restore power to any 4160 VAC safety-related bus. For the purpose of classification under this EAL, offsite power sources include any of the following:

- 345 KVAC system supplying power to the 13.8 KVAC system and the unit LVSATs
- Cross-tying with the opposite unit power supply
- Backfeeding power to the UATs through the 19 KVAC system and the main step-up transformer X-01. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source.

PBNP has the capability to cross-tie AC power from the other unit and therefore takes credit for the redundant power source for this IC. However, the inability to effect the cross-tie within 15

minutes warrants declaring a UE.~~Plants that have the capability to cross-tie AC power from a companion unit may take credit for the redundant power source in the associated EAL for this IC. Inability to effect the cross-tie within 15 minutes warrants declaring a NOUE.~~

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A Main Power Transformer Backfeed

SYSTEM MALFUNCTION

SU2

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

Inability to Reach Required Shutdown Within Technical Specification Limits.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SU2.1. Plant is not brought to required operating mode within ~~(site-specific)~~ Technical Specifications LCO Action Statement Time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the ~~site~~ PBNP Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate ~~NOUE~~ is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of a ~~NOUE~~ is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. PBNP Technical Specifications

SYSTEM MALFUNCTION

SU3

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN 15 Minutes

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SU3.1. UNPLANNED loss of most or all ~~(site-specific)~~ annunciators or indicators associated with the following safety systems for ~~greater than~~GREATER THAN 15 minutes

- ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that ~~most~~ plant designs provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This ~~will be~~is addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the ~~NO~~UE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

~~(Site-specific)~~The specified annunciators or indicators for this EAL ~~must~~ include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no IC is indicated during these modes of operation.

This ~~NO~~UE will be escalated to an Alert if a transient is in progress during the loss of annunciation or indication.

PBNP Basis Reference(s):

1. OM 1.1, Conduct of Plant Operations, Attachment 2

SYSTEM MALFUNCTION

SU4

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

Fuel Clad Degradation.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency Action Levels:~~Emergency Action Levels: (SU4.1 or SU4.2)

SU4.1. Failed Fuel Monitor (RE-109) GREATER THAN 750 mR/hr indicating fuel clad degradation~~(Site-specific) radiation monitor readings indicating fuel clad degradation~~ greater than Technical Specification allowable limits~~.~~

SU4.2. ~~(Site-specific) c~~Coolant sample activity GREATER THAN 50 µCi/gm dose equivalent I-131 ~~value~~ indicating fuel clad degradation greater than Technical Specification allowable limits.

Basis:

This IC is included as a ~~N~~OUE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. ~~EAL #1~~SU4.1 addresses ~~site-specific radiation monitor readings such as BWR air ejector monitors, PWR~~ the failed fuel monitor~~s~~, ~~etc.,~~ that provides indication of fuel clad integrity [Ref. 2]. 750 mR/hr is the value that corresponds to approximately 0.1% fuel clad damage.~~EAL #2~~ SU4.2 addresses coolant samples exceeding coolant technical specifications for iodine spike [Ref. 1]. Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs. ~~Though t~~The referenced Technical Specification limits are mode dependent.~~, it is appropriate that the EAL's be applicable in all modes, as they indicate a potential degradation in the level of safety of the plant. The companion IC to SU4 for the Cold Shutdown/Refueling modes is CU5.~~

PBNP Basis Reference(s):

1. Tech Spec 3.4.16 – RCS Specific Activity
2. Calc 2004-0019, Failed Fuel Monitor (RE-109) Reading / Fuel Damage Correlation

~~EPIP 10.2, Core Damage Estimation, Step 4.1~~

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

RCS Leakage.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency Action Levels:~~Emergency Action Levels: (SU5.1 or SU5.2)

SU5.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm.

SU5.2. Identified leakage GREATER THAN 25 gpm.

Basis:

This IC is included as a ~~N~~OUE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. TS 3.4.13, RCS Operational Leakage limits
2. OI 55, Primary Leak Rate Calculation
3. OM 3.19, Reactor Coolant System Leakage Determination

SYSTEM MALFUNCTION

SU6

Initiating Condition -- ~~NOTIFICATION OF~~ UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency Action Levels:~~Emergency Action Levels: (SU6.1 or SU6.2)

SU6.1. Loss of all Table C-1~~(site-specific list)~~ onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Plant Ppublic Aaddress Ssystem
• Security Radio
• Commercial ~~Telep~~Phone Ssystem
• Portable radios
• Sound power phones

SU6.2. Loss of all Table C-2~~(site-specific list)~~ offsite communications capability.

Table C-2 Offsite Communications Systems
• Emergency Notification System (ENS)
• Health Physics Network (HPN)
• Operations Control Counterpart Link (OCCL)
• Management Counterpart Link (MCL)
• Protective Measures Counterpart Link (PMCL)
• Reactor Safety Counterpart Link (RSCL)
• Nuclear Accident Reporting System (NARS)
• Commercial ~~Telep~~Phone System
• General Telephone Lines
• Manitowoc County Sheriff's Department Radio

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

~~Site-specific list for~~Table C-1 onsite communications loss [Ref. 1, 2] ~~must~~ encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system (Gaitronics) and radios / walkie talkies).

~~Site-specific list for~~Table C-2 offsite communications loss [Ref. 1, 2] ~~must~~ encompasses the loss of all means of communications with offsite authorities. This ~~should~~ includes the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

PBNP Basis Reference(s):

1. EPMP 2.1, Testing of Communications Equipment
2. EPMP 2.1A, Monthly Communications Test

Initiating Condition – ~~NOTIFICATION OF~~ UNUSUAL EVENT

Inadvertent Criticality.

OPERATING MODE APPLICABILITY Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level: (SU8.1) ~~or SU8.2)~~

~~1. An UNPLANNED extended positive period observed on nuclear instrumentation.~~

SU8.~~2~~1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses inadvertent criticality events. While the primary concern of this IC is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting an ~~NOUE~~ Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling IC is CU8.

This condition can be identified using the ~~period monitors/~~startup rate monitor. The term "~~extended~~sustained" is used in order to allow exclusion of expected short term positive ~~periods/~~startup rates from planned control rod movements ~~for PWRs and BWRs (~~such as shutdown bank withdrawal ~~for PWRs)~~. These short term positive ~~periods/~~startup rates are the result of the ~~increase~~ rise in neutron population due to subcritical multiplication.

This condition can be identified using startup rate monitors (1(2)NI-31D/32D - Source Range Startup Rate, and 1(2)NI-35D/36D - Intermediate Range Startup Rate) [Ref. 1].

Escalation would be by the Fission Product Barrier Matrix, as appropriate to the operating mode at the time of the event, or by Emergency Director Judgment.

Note: This EAL is SU8 following SU6. SU7 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

PBNP Basis Reference(s):

1. OP 1B, Reactor Startup

Initiating Condition -- ALERT

Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor ~~Scram~~Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual ~~Scram~~Trip Was Successful.

Operating Mode Applicability: Power Operation
Startup
Hot Standby

~~Example Emergency~~Emergency Action Level:

SA2.1. Indication(s) exist that a Reactor Protection System (RPS) setpoint was exceeded

AND

RPS automatic trip did NOT reduce power to LESS THAN 5%

AND

Any of the following operator actions are successful in reducing power to LESS THAN 5%:

- Use of Reactor Trip Buttons
- De-energizing 1(2)B01 and 1(2)B02

~~Indication(s) exist that indicate that reactor protection system setpoint was exceeded and automatic scram did not occur, and a successful manual scram occurred.~~

Basis:

This condition indicates failure of the automatic protection system to ~~scram~~ trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS integrity. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue. A manual ~~scram~~trip is any set of actions by the reactor operator(s) in the control room ~~at the reactor control console~~ which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip button~~, Alternate Rod Insertion~~). Failure of manual ~~scram~~trip would escalate the event to a Site Area Emergency.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees
2. BG-CSP-ST.0, Critical Safety Function Status Trees
3. EOP-0, Reactor Trip or Safety Injection

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SYSTEM MALFUNCTION

SA4

Initiating Condition – ALERT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SA4.1. UNPLANNED loss of most or all ~~(site-specific)~~ annunciators or indicators associated with the following safety systems for ~~greater than~~GREATER THAN 15 minutes

- -ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

AND

Either of the following: (a or b)

a. A SIGNIFICANT TRANSIENT is in progress.

OR

b. Compensatory non-alarming indications are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

SIGNIFICANT TRANSIENT includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Compensatory non-alarming indications include the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected.~~.~~ It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that ~~most~~ plant designs provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the ~~N~~OUE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

~~Site-specific~~The specified annunciators or indicators for this EAL ~~must~~ include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

~~"Compensatory non-alarming indications" in this context includes computer based information such as SPDS. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.~~

Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no IC is indicated during these modes of operation.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress.

Note: This EAL is SA4 following SA2. SA2 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

PBNP Basis Reference(s):

1. OM 1.1, Conduct of Plant Operations, Attachment 2

Initiating Condition -- ALERT

AC power capability to essential busses reduced to a single power source for GREATER THAN 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SA5.1. AC power capability to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06~~site-specific essential busses~~ reduced to only one of the following sources~~a single power source~~ for ~~greater than~~GREATER THAN 15 minutes

- A single emergency diesel generator (G01, G02, G03 or G04)
- LVSAT 1(2)-X04
- Cross-tying with the opposite unit power supply

AND

Any additional single failure will result in station blackout.

Basis:

This IC and the associated EALs are intended to provide an escalation from IC SU1, "Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes." The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency diesel generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency diesels with only one train of emergency busses being backfed from the unit main generator, or the loss of onsite emergency diesels with only one train of emergency busses being backfed from offsite power. Offsite power sources include the four 345 KVAC lines (111, 121, Q303 and 151) through the 13.8 KVAC system to the LVSAT and 345 KVAC backfed through the 19 KVAC system to the UAT. Note that the time required to effect a backfeed to the UAT is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source. Onsite power sources consist of the emergency diesel generators, the gas turbine generator (G05) feeding the 13.8 KVAC system to the LVSATs, the unit main turbine generator, and power supplied from the opposite unit. Several combinations of power failures could therefore satisfy this EAL. The subsequent loss of ~~this~~ the single remaining power source would escalate the event to a Site Area Emergency in accordance with IC SS1, "Loss of All Offsite and Loss of All Onsite AC Power to Essential Busses."

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A Unit 1, Main Power Transformer Backfeed

~~At multi-unit stations, the EALs should allow credit for operation of installed design features, such as cross-ties or swing diesels, provided that abnormal or emergency operating procedures address their use. However, these stations must also consider the impact of this condition on other shared safety functions in developing the site specific EAL.~~

SYSTEM MALFUNCTION

SS1

Initiating Condition -- SITE AREA EMERGENCY

Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SS1.1. Loss of all offsite power to safety-related 4160 VAC buses 1(2)-A05 AND 1(2)-A06 ~~power to (site-specific) transformers.~~

AND

Failure of all ~~(site-specific)~~ emergency generators to supply power to safety-related 4160 VAC buses~~emergency busses~~.

AND

Failure to restore power to at least one safety-related 4160 VAC ~~emergency~~ bus within 15 ~~(site-specific)~~ minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency. ~~The (site-specific) time duration should be selected to exclude transient or momentary power losses, but should not exceed 15 minutes.~~

Offsite power sources include the four 345 KVAC lines (111, 121, Q303 and 151) through the 13.8 KVAC system to the LVSAT and 345 KVAC backfed through the 19 KVAC system to the UAT. Note that the time required to effect a backfeed to the UAT is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source. Onsite power sources consist of the emergency diesel generators, the gas turbine generator (G05) feeding the 13.8 KVAC system to the LVSATs, the unit main turbine generator, and power supplied from the opposite unit.

Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power."

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 ~~essential~~ busses. Even though an safety-related 4160 VAC ~~essential~~ 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 this bus was the only energized bus then a Site Area Emergency per SS1 should be declared.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A U1, Main Power Transformer Backfeed

Initiating Condition -- SITE AREA EMERGENCY

Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor ~~Scram~~Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual ~~Scram~~Trip Was NOT Successful.

Operating Mode Applicability: Power Operation
Startup

~~Example Emergency~~Emergency Action Level:

SS2.1. ~~Indication(s) exist that automatic and manual scram were not successful.~~Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%.

Basis:

Automatic and manual ~~scram~~ trip are not considered successful if action away from the reactor control console was required to ~~scram~~ trip the reactor.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Emergency Director Judgment ICs.

Automatic or manual reactor trip is considered successful if actions taken at the main control panels (use of reactor trip buttons, de-energizing 1(2)B01 and 1(2)B02) result in reducing reactor power less than 5%.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees
2. BG-CSP-ST.0, CRITICAL SAF ETY FUNCTION STATUS TREES
3. CSP-S.1, Response to Nuclear Power Generation/ATWS
4. EOP-0, Reactor Trip or Safety Injection

SYSTEM MALFUNCTION

SS3

Initiating Condition -- SITE AREA EMERGENCY

Loss of All Vital DC Power.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SS3.1. Loss of aAll v~~V~~ital DC p~~P~~ower based on LESS THAN 115 VDC on 125 VDC buses D-01, D-02, D-03 and D-04~~(site-specific) bus voltage indications~~ for ~~greater than~~GREATER THAN 15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation, or Emergency Director Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The safety-related station batteries have been sized to carry their expected shutdown loads following a plant trip/LOCA and loss of offsite power or following a station blackout for a period of one hour without battery terminal voltage falling below the minimum bus voltages required for equipment operation. The bus voltage for full operability at all connected loads varies and is specific for each bus. At 115 VDC loss of equipment function would begin to occur. 105 VDC was not selected because it is the voltage of a completely discharged battery. By the time the bus voltage has degraded to below 105 VDC, the ability to monitor and control important plant safety functions would be severely compromised. 115 VDC represents a value at which some plant functions would begin to be lost. 115 VDC is below the minimum normal operating voltage, to preclude entering this EAL due to normal operational situations.

PBNP Basis Reference(s):

1. 0-SOP-DC-001/2/3/4

Initiating Condition -- SITE AREA EMERGENCY

Complete Loss of Heat Removal Capability.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SS4.1. Loss of core cooling (CSP-C.1) ~~and~~ AND heat sink (CSP-H.1) ~~(PWR)~~.

~~SS4.1. Heat Capacity Temperature Limit Curve exceeded (BWR).~~

Basis:

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs. ~~For BWRs the loss of heat removal function is indicated by the Heat Removal Capability Temperature Limit Curve being exceeded.~~

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to General Emergency would be via Abnormal Rad Levels / Radiological Effluent, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. ~~None~~CSP-C.1, Response to Inadequate Core Cooling
2. CSP-H.1, Response to Loss of Secondary Heat Sink

Initiating Condition -- SITE AREA EMERGENCY

Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SS6.1. ~~a.~~ Loss of most or all ~~(site-specific)~~ annunciators associated with the following safety systems

- ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

AND

SIGNIFICANT TRANSIENT in progress.

AND

~~b.~~ Compensatory non-alarming indications are unavailable.

AND

~~c.~~ Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, or maintain containment intact~~monitor (site-specific) safety functions~~ are unavailable.

~~**AND**~~

~~d. SIGNIFICANT TRANSIENT in progress.~~

Basis:

This IC and its associated EAL are intended to recognize the inability of the control room staff to monitor the plant response to a transient. A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public.

SIGNIFICANT TRANSIENT includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Compensatory non-alarming indications include the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room.
~~(Site-specific) annunciators for this EAL should be limited to include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., rad monitors, etc.)~~

~~"Compensatory non-alarming indications" in this context includes computer based information such as SPDS. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.~~

~~(Site-specific)~~ iIndications needed to monitor safety functions necessary for protection of the public ~~must~~ include control room indications, computer generated indications and dedicated annunciation capability. The specific indications ~~should be~~are those used to ~~determine such functions as~~monitor the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

"Planned" and "UNPLANNED" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification of the number of annunciators and indicators~~"Most"~~ is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

PBNP Basis Reference(s):

OM 1.1, Conduct of Plant Operations, Attachment 2

Initiating Condition -- GENERAL EMERGENCY

Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example Emergency~~Emergency Action Level:

SG1.1. Loss of all **offsite** power to ~~(site-specific) transformers~~safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06

AND

Failure of ~~(site-specific)~~ all emergency diesel generators to supply power to safety-related 4160 VAC buses~~emergency busses~~.

AND

Either of the following: (a or b)

a. Restoration of at least one safety-related 4160 VAC bus~~emergency bus~~ within ~~(site-specific)~~4 hours is ~~not~~ NOT likely

OR

b. ~~(Site-Specific) Indication of c~~Continuing degradation of core cooling based on Fission Product Barrier monitoring~~.~~ as indicated by conditions requiring entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. The ~~(site-specific)~~4 hours to restore AC power ~~can be~~is based on ~~a~~ the site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout,"~~ as available~~ [Ref 10]. Appropriate allowance for offsite emergency response including evacuation of surrounding areas should be considered. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.
The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.
In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent? (Refer to Table ~~s 3 and 4~~F-1 for more information.)

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

PBNP Basis Reference(s):

1. DBD-T-46, Section 3.1, Station Blackout
2. 10 CFR 50.63 and Regulatory Guide 1.155, Station Blackout
3. DBD-22, 4160 VAC System
4. DBD-18, 13.8 KVAC System
5. ECA 0.0, Loss of All AC Power
6. FSAR Section 8, Electrical Systems
7. AOP-14A U1, Main Power Transformer Backfeed
8. CSP-C.1, Response to Inadequate Core Cooling
9. CSP-C.2, Response to Degraded Core Cooling
10. FSAR Appendix A, "Station Blackout"

SG2

Initiating Condition -- GENERAL EMERGENCY

Failure of the Reactor Protection System to Complete an Automatic ~~Scram~~Trip and Manual ~~Scram~~Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability: Power Operation
Startup

~~Example Emergency~~Emergency Action Level:

SG2.1. Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%.

~~Indications exist that automatic and manual scram were not successful.~~

AND

Either of the following: (a or b)

a. Indication(s) exists that the core cooling is extremely challenged Core Cooling - RED path (CSP-C.1).

OR

b. Indication(s) exists that heat removal is extremely challenged ~~as indicated by conditions exist for entry into~~ Heat Sink - RED path (CSP-H.1).

Basis:

Automatic and manual ~~scram~~ trip are not considered successful if action away from the reactor control console is required to ~~scram~~ trip the reactor.

Under the conditions of this IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console, such as emergency boration ~~in PWRs, or standby liquid control in BWRs~~, the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence.

~~For PWRs, the extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F or that the reactor vessel water level is below the top of active fuel. For plants using CSFSTs, this EAL equates to a Core Cooling RED condition and an entry into function restoration procedure FR-S.1. For BWRs, the extreme challenge to the ability to cool the core is intended to mean that the reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level as described in the EOP bases.~~

~~Another consideration is the inability to initially remove heat during the early stages of this sequence. For PWRs, if emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist. For plants using CSFSTs, this EAL equates to a Heat Sink RED condition. For BWRs, considerations include inability to remove heat via the main condenser, or via the suppression pool or torus (e.g., due to high pool water temperature).~~

The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F or that the reactor vessel water level is below the top of active fuel. This EAL equates to a core cooling RED condition and an entry into a critical safety procedure (CSP-C.1).

Another consideration is the inability to initially remove heat during the early stages of this sequence. If emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist. This EAL equates to a Heat Sink RED condition and an entry into a critical safety procedure (CSP-H.1).

In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design (typically 3 to 5% power) a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees, Figures 1, 2 and 3
2. CSP-S.1, Response to Nuclear Power Generation/ATWS
3. CSP-C.1, Response to Inadequate Core Cooling
4. CSP-H.1, Response to Loss of Secondary Heat Sink

Enclosure IV

Attachment 2: Clean Technical Basis Document

Point Beach Nuclear Plant

Emergency Action Level Technical Basis Document

Emergency Action Level Technical Bases Document

Reviewed by;

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F - Fission Product Barrier Degradation F-1

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ACRONYMS

AC	Alternating Current
ATWS	Anticipated Transient Without Scram
CCW	Component Cooling Water
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
CMT	Containment
CSF	Critical Safety Function
CSFST	Critical Safety Function Status Tree
DC	Direct Current
DHR	Decay Heat Removal
DOT	Department of Transportation
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
EPIP	Emergency Plan Implementing Procedure
EPRI	Electric Power Research Institute
ERG	Emergency Response Guideline
ESF	Engineered Safeguards Feature
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
GE	General Emergency
HPSI	High Pressure Safety Injection
IC	Initiating Condition
IDLH	Immediately Dangerous to Life and Health
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI	Independent Spent Fuel Storage Installation
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LFL	Lower Flammability Limit

LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection
MSIV	Main Steam Isolation Valve
mR	milliRoentgen
mrem	milliRem
Mw	Megawatt
NEI	Nuclear Energy Institute
NESP	National Environmental Studies Project
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
PSIG	Pounds per Square Inch Gauge
R	Roentgen
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specifications
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RVLIS	Reactor Vessel Level Indicating System
SAE	Site Area Emergency
SG	Steam Generator
SI	Safety Injection
SPDS	Safety Parameter Display System
SRO	Senior Reactor Operator
SSE	Safe Shutdown Earthquake
TEDE	Total Effective Dose Equivalent
TOAF	Top of Active Fuel
TSC	Technical Support Center
UE	Notification Of Unusual Event
WE	Westinghouse Electric
WOG	Westinghouse Owners Group

1. PURPOSE

This document provides the detailed set of Emergency Action Levels (EALs) applicable to the Point Beach Nuclear Plant (PBNP) and the associated Technical Bases using the EAL development methodology found in NEI 99-01 Revision 4 [Ref. 2.1]. Personnel responsible for implementation of EPIP-1.2 (Emergency Classification) [Ref. 2.2], and the Emergency Action Level Matrix [Ref. 2.3] may use this document as a technical reference and an aid in EAL interpretation.

2. REFERENCES

2.1 NEI 99-01 Revision 4, Methodology for Development of Emergency Action Levels

2.2 EPIP-1.2 (Emergency Classification)

2.3 Emergency Action Level Matrix

2.4 PBNP Technical Specifications Table 1.1-1

3. DISCUSSION

3.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the PBNP Emergency Plan.

In 1992, the NRC endorsed NUMARC/NESP-007 "Methodology for Development of Emergency Action Levels" as an alternative to NUREG 0654 EAL guidance.

NEI 99-01 (NUMARC/NESP-007) Revision 4 represents the most recently NRC endorsed methodology per RG 1.101 Rev 4, "Emergency Planning and Preparedness for Nuclear Power Reactors." Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.
- Addressing initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations.
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Using NEI 99-01 Rev. 4, PBNP conducted an EAL implementation upgrade project that produced the EALs discussed herein. While the upgraded EALs are site-specific, an objective of the project was to ensure to the extent possible EAL conformity and consistency between the NMC plant sites.

3.2 Key Definitions in EAL Methodology

The following definitions apply to the generic EAL methodology:

EMERGENCY CLASS: One of a minimum set of names or titles, established by the Nuclear Regulatory Commission (NRC), for grouping of normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time sensitive onsite and off site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called:

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

Section 3.3 provides further discussion of the emergency classes.

INITIATING CONDITION (IC): One of a predetermined subset of nuclear power plant conditions when either the potential exists for a radiological emergency, or such an emergency has occurred.

- An IC is an emergency condition which sets it apart from the broad class of conditions that may or may not have the potential to escalate into a radiological emergency.
- It can be a continuous, measurable function that is outside technical specifications, such as elevated RCS temperature or falling reactor coolant level (a symptom).
- It also encompasses occurrences such as FIRE (an event) or reactor coolant pipe failure (an event or a barrier breach).

EMERGENCY ACTION LEVEL (EAL): A pre determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

- There are times when an EAL will be a threshold point on a measurable continuous function, such as a primary system coolant leak that has exceeded technical specifications.
- At other times, the EAL and the IC will coincide, both identified by a discrete event that places the plant in a particular emergency class.

3.3 Recognition Categories

ICs and EALs are grouped in one of several categories. This classification scheme incorporates symptom-based, event-based, and barrier-based ICs and EALs.

- R - Abnormal Rad Levels/Radiological Effluent
- C - Cold Shutdown./ Refueling System Malfunction
- E - Independent Spent Fuel Storage Installation (ISFSI)
- F - Fission Product Barrier Degradation
- H - Hazards
- S - System Malfunction

Some recognition categories are further divided into one or more subcategories depending on the types and number of plant conditions that dictate emergency classifications. An EAL may or may not exist for each subcategory at all four classification levels. Similarly, more than one EAL may exist for a subcategory in a given emergency classification when appropriate (i.e., no EAL at the General Emergency level but three EALs at the Unusual Event level).

ICs and EALs can be grouped in one of several schemes. This generic classification scheme incorporates symptom-based, event-based, and barrier-based ICs and EALs.

The symptom-based category for ICs and EALs refers to those indicators that are measurable over some continuous spectrum, such as core temperature, coolant levels, containment pressure, etc. When one or more of these indicators begin to show off-normal readings, reactor operators are trained to identify the probable causes and potential consequences of these "symptoms" and take corrective action. The level of seriousness indicated by these symptoms depends on the degree to which they have exceeded technical specifications, the other symptoms or events that are occurring contemporaneously, and the capability of the licensed operators to gain control and bring the indicator back to safe levels.

Event-based EALs and ICs refer to occurrences with potential safety significance, such as the failure of a high-pressure safety injection pump, a safety valve failure, or a loss of electric power to some part of the plant. The range of seriousness of these "events" is dependent on the location, number of contemporaneous events, remaining plant safety margin, etc.

Barrier-based EALs and ICs refer to the level of challenge to principal barriers used to assure containment of radioactive materials contained within a nuclear power plant. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers concurrently under challenge. In reality, barrier-based EALs are a subset of symptom-based EALs that deal with symptoms indicating fission product barrier

challenges. These barrier-based EALs are primarily derived from Emergency Operating Procedure (EOP) Critical Safety Function (CSF) Status Tree Monitoring (or their equivalent). Challenge to one or more barriers generally is initially identified through instrument readings and periodic sampling. Under present barrier-based EALs, deterioration of the reactor coolant system pressure boundary or the fuel clad barrier usually indicates an "Alert" condition, two barriers under challenge a Site Area Emergency, and loss of two barriers with the third barrier under challenge is a General Emergency. The fission product barrier matrix described in Section 5-F is a hybrid approach that recognizes that some events may represent a challenge to more than one barrier, and that the containment barrier is weighted less than the reactor coolant system pressure boundary and the fuel clad barriers.

Symptom-based ICs and EALs are most easily identified when the plant is in a normal startup, operating or hot shutdown mode of operation, with all of the barriers in place and the plant's instrumentation and emergency safeguards features fully operational as required by technical specifications. It is under these circumstances that the operations staff has the most direct information of the plant's systems, displayed in the main control room. As the plant moves through the decay heat removal process toward cold shutdown and refueling, barriers to fission products are reduced (i.e., reactor coolant system pressure boundary may be open) and fewer of the safety systems required for power operation are required to be fully operational. Under these plant operating modes, the identification of an IC in the plant's operating and safety systems becomes more event-based, as the instrumentation to detect symptoms of a developing problem may not be fully effective; and engineered safeguards systems, such as the Emergency Core Cooling System (ECCS), are partially disabled as permitted by the plant's Technical Specifications.

Barrier-based ICs and EALs also are heavily dependent on the ability to monitor instruments that indicate the condition of plant operating and safety systems. Fuel cladding integrity and reactor coolant levels can be monitored through several indicators when the plant is in a normal operating mode, but this capability is much more limited when the plant is in a refueling mode, when many of these indicators are disconnected or off-scale. The need for this instrumentation is lessened, however, and alternate instrumentation is placed in service when the plant is shut down.

It is important to note that in some operating modes there may not be definitive and unambiguous indicators of containment integrity available to control room personnel. For this reason, barrier-based EALs should not place undue reliance on assessments of containment integrity in all operating modes. Generally, Technical Specifications relax maintaining containment integrity requirements in modes 5 and 6 in order to provide flexibility in performance of specific tasks during shutdown conditions. Containment pressure and temperature indications may not increase if there is a pre-existing breach of containment integrity. At most plants, a large portion of the containment's exterior cannot be monitored for leakage by radiation monitors.

Several categories of emergencies have no instrumentation to indicate a developing problem, or the event may be identified before any other indications are recognized. A reactor coolant pipe could break; FIRE alarms could sound; radioactive materials could be released; and any number of other events can occur that would place the plant in an

emergency condition with little warning. For emergencies related to the reactor system and safety systems, the ICs shift to an event based scheme as the plant mode moves toward cold shutdown and refueling modes. For non-radiological events, such as FIRE, external floods, wind loads, etc., as described in NUREG-0654 Appendix 1, event-based ICs are the norm.

In many cases, a combination of symptom-, event- and barrier-based ICs will be present as an emergency develops. In a loss of coolant accident (LOCA), for example:

- Coolant level is dropping; (symptom)
- There is a leak of some magnitude in the system (pipe break, safety valve stuck open) that exceeds plant capabilities to make up the loss; (barrier breach or event)
- Core (coolant) temperature is rising; (symptom) and
- At some level, fuel failure begins with indicators such as high off-gas, high coolant activity samples, etc. (barrier breach or symptom)

To the extent possible, the EALs are symptom based. That is, the action level is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. But, a purely symptom based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

Category R - Abnormal Rad Levels/Radiological Effluent and Category F - Fission Product Barrier Degradation are primarily symptom-based. The symptoms are indicative of actual or potential degradation of either fission product barriers or personnel safety.

Other categories tend to be event-based. For example, System Malfunctions are abnormal and emergency events associated with vital plant system failures, while Hazards are those non-plant system related events that have affected or may affect plant safety.

3.4 Emergency Class Descriptions

There are three considerations related to the emergency classes. These are:

- The potential impact on radiological safety, either as now known or as can be reasonably projected.
- How far the plant is beyond its predefined design, safety and operating envelopes.
- Whether or not conditions that threaten health are expected to be confined to within the site boundary.

The ICs deal explicitly with radiological safety affect by escalating from levels corresponding to releases within regulatory limits to releases beyond EPA Protective Action Guideline (PAG) plume exposure levels.

UNUSUAL EVENT: Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

- Potential degradation of the level of safety of the plant is indicated primarily by exceeding plant technical specification Limiting Condition of Operation (LCO) allowable action statement time for achieving required mode change.

- Precursors of more serious events may be included because precursors represent a potential degradation in the level of safety of the plant.

- Minor releases of radioactive materials are included. In this emergency class, however, releases do not require monitoring or offsite response (e.g., dose consequences of less than 10 millirem).

ALERT: Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

- Rather than discussing the distinguishing features of "potential degradation" and "potential substantial degradation," a comparative approach would be to determine whether increased monitoring of plant functions is warranted at the Alert level as a result of safety system degradation. This addresses the operations staff's need for help, independent of whether an actual decrease in plant safety is determined. This increased monitoring can then be used to better determine the actual plant safety state, whether escalation to a higher emergency class is warranted, or whether de-escalation or termination of the emergency class declaration is warranted. Dose consequences from these events are small fractions of the EPA PAG plume exposure levels, i.e., about 10 millirem to 100 millirem TEDE.

SITE AREA EMERGENCY: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for 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.

- The discriminator (threshold) between Site Area Emergency and General Emergency is whether or not the EPA PAG plume exposure levels are expected to be exceeded outside the site boundary.

- This threshold, in addition to dynamic dose assessment considerations discussed in the EAL guidelines, clearly addresses NRC and offsite emergency response agency concerns as to timely declaration of a General Emergency.

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. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

- The bottom line for the General Emergency is whether evacuation or sheltering of the general public is indicated based on EPA PAGs and, therefore, should be interpreted to include radionuclide release regardless of cause.

- To better assure timely notification, EALs in this category are primarily expressed in terms of plant function status, with secondary reliance on dose projection. In terms of fission product barriers, loss of two barriers with loss or potential loss of the third barrier constitutes a General Emergency.

3.5 Emergency Class Thresholds

The most common bases for establishing these boundaries are the technical specifications and setpoints for each plant that have been developed in the design basis calculations and the Final Safety Analysis Report (FSAR).

For those conditions that are easily measurable and instrumented, the boundary is likely to be the EAL (observable by plant staff, instrument reading, alarm setpoint, etc.) that indicates entry into a particular emergency class. For example, the main steam line radiation monitor may detect high radiation that triggers an alarm. That radiation level also may be the setpoint that closes the main steam isolation valves (MSIV) and initiates the reactor scram. This same radiation level threshold, depending on plant-specific parameters, also may be the appropriate EAL for a direct entry into an emergency class.

In addition to the continuously measurable indicators, such as coolant temperature, coolant levels, leak rates, containment pressure, etc., the FSAR provides indications of the consequences associated with design basis events. Examples would include steam pipe breaks, MSIV malfunctions, and other anticipated events that, upon occurrence, place the plant immediately into an emergency class.

Another approach for defining these boundaries is the use of a plant-specific probabilistic safety assessment (PSA - also known as probabilistic risk assessment, PRA). PSAs have been completed for several individual plants, but this is by no means comprehensive. There are, however, PSAs that have been completed for representative plant types such as is done in NUREG-1150, "Severe Accident Risks: An Assessment for Five Nuclear Power Plants," as well as several other utility-sponsored PSAs. Existing PSAs can be used as a good first approximation of the relevant ICs and risk associated with emergency conditions for existing plants.

Another critical element of the analysis to arrive at these threshold (boundary) conditions is the time that the plant might stay in that condition before moving to a higher emergency class. In particular, station blackout coping analyses performed in response to 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout," may be used to determine whether a specific plant enters a Site Area Emergency or a General Emergency directly, and when escalation to General Emergency is indicated. The time dimension is critical to the EAL since the purpose of the emergency class for state and local officials is to notify them of

the level of mobilization that may be necessary to handle the emergency. This is particularly true when a "Site Area Emergency" or "General Emergency" is imminent. Establishing EALs for such conditions must take estimated evacuation time into consideration to minimize the potential for the plume to pass while evacuation is underway.

Regardless of whether or not containment integrity is challenged, it is possible for significant radioactive inventory within containment to result in EPA PAG plume exposure levels being exceeded even assuming containment is within technical specification allowable leakage rates. With or without containment challenge, however, a major release of radioactivity requiring offsite protection actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

3.6 Emergency Action Levels

With the emergency classes defined, the thresholds that must be met for each EAL to be placed under the emergency class can be determined. There are two basic approaches to determining these EALs. EALs and emergency class boundaries coincide for those continuously measurable, instrumented ICs, such as radioactivity, core temperature, coolant levels, etc. For these ICs, the EAL will be the threshold reading that most closely corresponds to the emergency class description using the best available information.

For discrete (discontinuous) events, the approach will have to be somewhat different. Typically, in this category are internal and external hazards such as FIRE or earthquake. The purpose for including hazards in EALs is to assure that station personnel and offsite emergency response organizations are prepared to deal with consequential damage these hazards may cause. If, indeed, hazards have caused damage to safety functions or fission product barriers, this should be confirmed by symptoms or by observation of such failures. Therefore, it may be appropriate to enter an Alert status for events approaching or exceeding design basis limits such as Operating Basis Earthquake, design basis wind loads, FIRE within VITAL AREAs, etc. This would give the operating staff additional support and improved ability to determine the extent of plant damage. if damage to barriers or challenges to Critical Safety Functions (CSFs) have occurred or are identified, then the additional support can be used to escalate or terminate the Emergency Class based on what has been found. Of course, security events must reflect potential for increasing security threat levels.

Plant emergency operating procedures (EOPs) are designed to maintain and/or restore a set of CSFs which are listed in the order of priority for restoration efforts during accident conditions. While the actual nomenclature of the CSFs may vary among plants, generally the PWR CSF set includes:

- Subcriticality
- Core cooling
- Heat sink
- Pressure-temperature-stress (RCS integrity)

- Containment
- RCS inventory

There are diverse and redundant plant systems to support each CSF. By monitoring the CSFs instead of the individual system component status, the impact of multiple events is inherently addressed, e.g., the number of operable components available to maintain the critical safety function.

The EOPs contain detailed instructions regarding the monitoring of these functions and provides a scheme for classifying the significance of the challenge to the functions. In providing EALs based on these schemes, the emergency classification can flow from the EOP assessment rather than being based on a separate EAL assessment. This is desirable as it reduces ambiguity and reduces the time necessary to classify the event.

As an example, consider that the Westinghouse Owner's Group (WOG) Emergency Response Guidelines (ERGs) classify challenges as YELLOW, ORANGE, and RED paths. If the core exit thermocouples exceed 1200 degrees F or 700 degrees F with low reactor vessel water level, a RED path condition exists. The ERG considers a RED path as "... an extreme challenge to a plant function necessary for the protection of the public ..." This is almost identical to the present NRC NUREG-0654 description of a site area emergency "... actual or likely failures of plant functions needed for the protection of the public ..." It reasonably follows that if any CSF enters a RED path, a site area emergency exists. A general emergency could be considered to exist if core cooling CSF is in a RED path and the EOP function restoration procedures have not been successful in restoring core cooling.

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds has been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

3.7 Treatment of Multiple Events And Emergency Class Upgrading

The above discussion deals primarily with simpler emergencies and events that may not escalate rapidly. However, usable EAL guidance must also consider rapidly evolving and complex events. Hence, emergency class upgrading and consideration of multiple events must be addressed.

There are three approaches presently in use for covering multiple events and emergency class upgrading. These approaches are:

(U1) Multiple contemporaneous events are counted and are the basis for escalating to a higher emergency class. For example, two or more contemporaneous Alerts escalate to a Site Area Emergency.

(U2) The emergency class is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency.

(U3) Emergency Director judgment. Although all emergency classifications require judgment, some utilities rely on Emergency Director judgment with little or no additional explicit guidance.

An additional approach for plants with PRAs is to make use of event tree analysis to define combinations of events which lead to equivalent risks. Such event sequences should have an equal emergency classification assigned. However, the chief drawback to this approach as well as (U1) above, is that multiple events may be masked when they actually occur. Further, for plants using symptom-based (and barrier-based) emergency procedures, direct perception of multiple events is unnecessary.

Emergency class upgrading for multi-unit stations with shared safety-related systems and functions must also consider the effects of a loss of a common system on more than one unit (e.g. potential for radioactive release from more than one core at the same site). For example, many two-unit stations have their control panels for both units in close proximity within the same room. Thus, control room evacuation most likely would affect both units. There are a number of other systems and functions which may be shared at a given multi-unit station. This must be considered in the emergency class declaration and in the development of appropriate site-specific ICs and EALs based on the generic EAL guidance.

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds has been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

PBNP will utilize the following methodology:

With appropriate consideration for Emergency Director judgment EALs, properly structured EALs on a fission product barrier basis and which include equivalent risk, will appropriately escalate multiple events to a higher emergency class. For example, common cause failures such as loss of ultimate heat sink or loss of all AC power, will result in multiple contemporaneous symptoms indicating safety system functional failures and increasing challenge to fission product barriers. It is the existence of these symptoms (barrier challenges) that escalate the emergency class, whether there are one or multiple causes

The emergency class is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency.

Since PBNP is a dual-unit plant, emergency class upgrading must consider the effects of a loss of a common system on the other unit. For example, the control panels for both units share the same room. Thus, control room evacuation most likely would affect both units. There are a number of other systems and functions which may be shared. This must be considered in the emergency class declaration.

3.8 Emergency Class Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing. There are several approaches presently in use for emergency class downgrading. These approaches are:

(D1) Terminate the emergency class declaration.

(D2) Recovery from emergency class.

(D3) Combination of downgrading approaches. Many utilities reviewed include the option to downgrade to a lower emergency class. This is consistent with actions called for in NUREG-0654 Appendix 1. However, these utilities state that their experience more closely resembles (D1) and (D2) above as practical choices.

Another approach possible with risk-based EALs is a relatively simple approach for upgrading to a higher emergency class when the risk increases and downgrading when risk decreases. The boundaries for emergency categories are defined in terms of risk in this approach, and discrete events fall into these categories based on risk. This means that within each emergency class, there is uniformity to the relative levels of risk to human health and safety from radiological accidents. However, this option may not be practical when applied to actual emergencies, especially those involving General Emergencies.

PBNP will utilize the following methodology:

> **A combination approach involving recovery from General Emergencies and Site Area Emergencies and termination from UEs, Alerts, causing no long-term plant damage appears to be the best choice. Downgrading to lower emergency classes adds notifications but may have merit under certain circumstances.**

3.9 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. For example, an emergency classification is warranted when automatic and manual actions taken within the control room do not result in a required reactor trip. However, it is likely that actions taken outside of the control room will be successful, probably before the Emergency Director classifies the event. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses (e.g., coolant sampling, may be necessary).

In general, observe the following guidance: Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event

and other termination criteria are met. For example, a momentary event, such as an ATWS or an earthquake, requires declaration even though the condition may have been resolved by the time the declaration is made.

- An ATWS represents a failure of a front line safety system (RPS) designed to protect the health and safety of the public.
- The affect of an earthquake on plant equipment and structures may not be readily apparent until investigations are conducted.

There may be cases in which a plant condition that exceeded an EAL threshold was not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Rev. 1, Section 3 should be applied.

3.10 Operating Mode Applicability

Technical Specifications [Ref. 2.4] provides definitions for the following operating modes:

1 Power Operations

K_{eff} is GREATER THAN OR EQUAL TO 0.99 and reactor power is GREATER THAN 5% rated thermal power.

2 Startup

K_{eff} is GREATER THAN OR EQUAL TO 0.99 and reactor power is LESS THAN OR EQUAL TO 5% rated thermal power.

3 Hot Standby

K_{eff} is LESS THAN 0.99 and average reactor coolant temperature (Tavg) GREATER THAN OR EQUAL TO 350°F.

4 Hot Shutdown

K_{eff} is LESS THAN 0.99 and average reactor coolant temperature (Tavg) LESS THAN 350°F and GREATER THAN 200°F.

5 Cold Shutdown

K_{eff} is LESS THAN 0.99 and average reactor coolant temperature (Tavg) LESS THAN OR EQUAL TO 200°F.

6 Refuel

One or more vessel head closure bolts less than fully tensioned

In addition to the Technical Specification operating modes, NEI 99-01 [Ref. 1] defines the following additional mode:

D <u>Defueled</u>

All reactor fuel removed from Reactor Vessel (full core off load during refueling or extended outage)

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action is initiated in response to the condition) should be compared to the mode applicability of the EALs. If a lower or higher plant operating mode is reached before the emergency classification is made, the declaration shall be based on the mode that existed at the time the event occurred.

Recognition categories are associated with the operating modes listed in the following matrix:

	Recognition Category					
Mode	R	C	E	F	H	S
1 - Power Operations	X			X	X	X
2 - Startup	X			X	X	X
3 - Hot Standby	X			X	X	X
4 - Hot Shutdown	X			X	X	X
5 - Cold Shutdown	X	X			X	
6 - Refuel	X	X			X	
D - Defueled	X	X			X	
N/A			X			

3.11 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon loss of or potential loss to one or more of the three fission product barriers. "Loss" and "potential loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials and "potential loss" means imminent loss of the barrier.

The primary fission product barriers are:

- Fuel Cladding (FC): Zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the FC barrier.

- Reactor Coolant System (RCS): The reactor vessel shell, vessel head, vessel nozzles and penetrations and all primary systems directly connected to the reactor vessel up to the first containment isolation valve comprise the RCS barrier.

- Containment (CMT): The vapor containment structure and all isolation valves required to maintain containment integrity under accident conditions comprise the Containment barrier.

3.12 Emergency Classification Based on Fission Product Barrier Degradation

The following criteria are the bases for event classification related to fission product barrier loss or challenge:

- Unusual Event:

  Any loss or any potential loss of Containment

- Alert:

  Any loss or any potential loss of either Fuel Cladding or RCS

- Site Area Emergency:

  Loss or potential loss of any two barriers

- General Emergency:

  Loss of any two barriers and loss or potential loss of third barrier

3.13 EAL Relationship to EOPs and Critical Safety Function Status

Where possible, the EALs have been made consistent with and utilize the conditions defined in the PBNP Critical Safety Function Status Trees (CSFSTs). While the symptoms that drive operator actions specified in the CSFSTs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. Where these symptoms are clearly representative of one of the NEI Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the CSFSTs, classification of emergencies using these EALs is not dependent upon Emergency Operating Procedures (EOP) entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

3.14 Imminent EAL Thresholds

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the thresholds has been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes. Explicit EALs, specifying use of Emergency Director judgment, are given in the Hazards, ISFSI and Fission Product Barrier Degradation categories.

4. TECHNICAL BASES INFORMATION

4.1 Recognition Category Organization

The technical bases of the EALs are provided under Recognition Categories R, C, E, F, H and S of this document. A table summarizing the Initiating Conditions introduces each category. The tables provide an overview of how the ICs are related under each emergency class. ICs within each category are listed according to classification (as applicable) in the following order: Unusual Event, Alert, Site Area Emergency, and General Emergency.

For Recognition Category F, Table F-0 defines the emergency classifications associated with barrier loss and potential loss. Table F-1 lists the thresholds associated with the loss and potential loss of each fission product barrier. The presentation method shown for Table F-1 was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. Basis discussion of the thresholds immediately follows Table F-1.

4.2 Initiating Condition Structure

ICs in Recognition Categories R, C, E, H and S are structured in the following manner:

- Recognition Category Title
- IC Identifier:
    - First character identifies the category by letter (R, C, E, H and S)
    - Second character identifies the emergency classification level (UE for Unusual Event, A for Alert, S for Site Area Emergency, and G for General Emergency)
    - Third character is the numerical sequence as given in Revision 4 of NEI 99-01 [Ref. 1] (e.g., SA2). Due to document revisions, certain NEI ICs have been deleted, leaving gaps in the numerical sequence.

- Emergency Class: Unusual Event, Alert, Site Area Emergency, or General Emergency

- IC Description

- Operating Mode Applicability: Refers to the operating mode during which the IC/EAL is applicable

- Emergency Action Level(s): EALs are the conditions applicable to the criteria of the IC and are used to determine the need to classify an event/condition. If more than one EAL is applicable to an IC, emergency classification is required when any EAL within the IC reaches the EAL threshold. To clarify this intent, ICs with multiple EALs include a parenthetical phrase in the EAL title line, indicating that each constitutes an emergency classification. For example, the phrase "(RA1.1 or RA1.2)" indicates that either EAL is An Alert.

- Basis: Provides information that explains the IC and EAL(s). Plant source document references are provided as needed to substantiate site-specific information included in the EALs and bases.

4.3 EAL Identification

The EAL identifier is the IC identifier followed by a period and sequence number (e.g., RU1.1, RU1.2, etc.). If only one EAL is assigned to an IC, the EAL is given the number one.

The primary purpose of the EAL identifier is to uniquely distinguish each classifiable condition. Secondary purposes are to assist location of an EAL within the EAL classification scheme and to announce the emergency classification level.

5. EMERGENCY ACTION LEVEL BASES

The information is presented by Recognition Categories:

R - Abnormal Rad Levels/Radiological Effluent

- Radiological Effluents
- Abnormal Radiation Levels
- Irradiated Fuel Accidents

C - Cold Shutdown / Refueling System Malfunction

- Loss of AC Power
- Loss of DC Power
- Decay Heat Removal
- RCS Leakage/RCS Draindown
- Loss of Reactor Vessel Inventory
- Fuel Clad Degradation

- Loss of Communications
- Inadvertent Criticality

E - Independent Spent Fuel Storage Installation (ISFSI)

- Dry Fuel Storage

F - Fission Product Barrier Degradation

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

H – Hazards

- Security Events
- Control Room Evacuation
- Natural or man-Made Events
- FIRE/EXPLOSION
- Toxic or Flammable Gases
- Discretionary

S - System Malfunction

- Loss of AC Power
- Loss of DC Power
- Failure of Reactor Protection System (RPS)
- Decay Heat Removal
- Loss of Annunciators
- RCS Leakage
- Fuel Clad Degradation
- Loss of Communications
- Technical Specifications
- Inadvertent Criticality

The Initiating Conditions for each of the above Recognition Categories R, C, E, F, H, and S are in the order of UE, Alert, Site Area Emergency, and General Emergency. For all Recognition Categories, an Initiating Condition matrix versus Emergency Class is first shown. For Recognition Category F, the barrier-based EALs are presented in Table F-1. For all other Recognition Categories Initiating Condition matrices are not required. The

purpose of the IC matrices is to provide the reader with an overview of how the ICs are logically related under each Emergency Class.

Each of the EAL guides in Recognition Categories R, C, E, F, H, and S is structured in the following way:

- **Recognition Category** - As described above.

- **Emergency Class** - UE, Alert, Site Area Emergency or General Emergency.

- **Initiating Condition** – Symptom- or Event-Based, Generic Identification and Title.

- **Operating Mode Applicability** - refers to the operating mode (PWRs) during which the IC/EAL is applicable - Power Operation (includes Startup Mode in PWRs), Hot Standby (includes Hot Standby / Startup Condition in BWRs), Hot Shutdown, Cold Shutdown, Refueling, Defueled, All, or None. These modes are defined in each licensee's technical specifications. The mode classifications and terminology appropriate to the specific facility should be used. Note that Permanently Defueled and ISFSI IC/EALs have no mode applicability.
  If an IC or EAL includes an explicit reference to a technical specification, and the technical specification is not applicable because of operating mode, then that particular IC or EAL is also not applicable.

- **Example Emergency Action Level(s)** – these EALs are examples of conditions and indications that were considered to meet the criteria of the IC. These examples were not intended to be all encompassing, and some may not apply to a particular facility. Utilities should generally address each example EAL that applies to their site. If an example EAL does not apply because of its wording, e.g., specifies instrumentation not available at the site, the utility should identify other available means for entry into the IC. Ideally, the example EALs used will be unambiguous, expressed in site-specific nomenclature, and be readily discernible from control room instrumentation.

- **Basis** – provides information that explains the IC and example EALs. The bases are written to assist the personnel implementing the generic guidance into site-specific procedures. Site-specific deviations from the IC/EALs should be compared to the Basis for that IC to ensure that the fundamental intent of each IC/EAL is met. Some bases provide information intended to assist with establishing site-specific instrumentation values.

For Recognition Category F, basis information is presented in a format consistent with NEI 99-01 Rev 4. Tables 3 and 4. The presentation method shown for Fission Product Barrier Function Matrix was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. Other acceptable methods of achieving these goals which are currently in use include flow charts, block diagrams, and checklist tables. Utilities selecting these alternative need to ensure that all possible EAL combinations in the Fission Product Barrier Function Matrix are addressed in their presentation method.

For a number of Alerts, IC/EALs are chosen based on hazards which may cause damage to plant safety functions (i.e., tornadoes, hurricanes, FIRE in plant VITAL AREAs) or require additional help directly (control room evacuation) and thus increased monitoring of the plant is warranted. The symptom-based and barrier-based IC/EALs are sufficiently anticipatory to address the results of multiple failures, regardless of whether there is or is not a common cause. Declaration of the Alert will already result in the manning of the TSC for assistance and additional monitoring. Thus, direct escalation to the Site Area Emergency is unnecessary. Other Alerts, that have been specified, correspond to conditions which are consistent with the emergency class description.

The basis for declaring a Site Area Emergency and General Emergency is primarily the extent and severity of fission product barrier challenges, based on plant conditions as presently known or as can be reasonably projected.

With regard to the Hazards Recognition Category, the existence of a hazard that represents a potential degradation in the level of safety of the plant is the basis of NOUE classification. If the hazard results in VISIBLE DAMAGE to plant structures or equipment associated with safety systems or if system performance is affected, the event may be escalated to an Alert. The reference to "duration" or to "damage" to safety systems is intended only to size the event. Consequential damage from such hazards, if observed, would be the basis for escalation to Site Area Emergency or General Emergency, by entry to System Malfunction or Fission Product Barrier IC/EALs.

Basis Documents are attached as follows:

R- Abnormal Rad Levels / Radiological Effluent	R-1
C- Cold Shutdown / Refueling System Malfunction	C-1
E – Independent Fuel Storage Installation (USFSI)	E-1
F – Fission Product Barrier Degradation	F-1
H – Hazards	H-1
S- System Malfunction	S-1

6. DEFINITIONS

In the ICs and EALs, selected words are in uppercase print. These words are defined terms. Definitions are provided below.

A

ACTUATE: To put into operation; to move to action; commonly used to refer to automated, multi-faceted operations. "Actuate ECCS"

ADVERSARY: 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.

AFFECTING SAFE SHUTDOWN: event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable HOT or COLD

SHUTDOWN condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in HOT SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is not "AFFECTING SAFE SHUTDOWN."

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in COLD SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is "AFFECTING SAFE SHUTDOWN."

ALERT: Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guide exposure levels.

AVAILABLE: The state or condition of being ready and able to be used (placed into operation) to accomplish the stated (or implied) action or function. As applied to a system, this requires the operability of necessary support systems (electrical power supplies, cooling water, lubrication, etc.).

B

BOMB refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

C

CIVIL DISTURBANCE is a group of two or more persons violently protesting station operations or activities at the site.

CLOSE: To position a valve or damper so as to prevent flow of the process fluid. To make an electrical connection to supply power.

CONFINEMENT BOUNDARY is the barrier(s) between areas containing radioactive substances and the environment.

CONFIRM / CONFIRMATION: To validate, through visual observation or physical inspection, that an assumed condition is as expected or required, without taking action to alter the "As found" configuration.

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.

CONTIGUOUS: Being in actual contact; touching along a boundary or at a point

CONTROL: Take action, as necessary, to maintain the value of a specified parameter within applicable limits; to fix or adjust the time, amount, or rate of; to regulate or restrict.

D

DEVIATION: Instances where the guidance reference IC/EAL (99-01) differs in wording from proposed revision and is altered in intent, such that classification of event could be specifically different between guidance reference and licensee proposed EAL.

DIFFERENCE: Instances where the guidance reference IC/EAL (99-01) differ in wording but agree in meaning and intent.

DISCHARGE: Removal of a fluid/gas from a volume or system.

E

ENTER: To go into.

ESTABLISH: To perform actions necessary to meet a stated condition. "Establish communication with the Control Room."

EVACUATE: To remove the contents of; to remove personnel from an area.

EXCEEDS: To go or be beyond a stated or implied limit, measure, or degree.

EXIST: To have being with respect to understood limitations or conditions.

EXPLOSION is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

EXTORTION is an attempt to cause an action at the station by threat of force.

F

FAILURE: A state of inability to perform a normal function.

FAULTED: in a steam generator, the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE is 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.

FISSION PRODUCT BARRIERS (FPB): Multiple physical barriers any of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The FPBs are the Reactor Fuel Cladding (FC), Reactor Coolant System (RCS) and Containment (PC).

G

GENERAL EMERGENCY: Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed Protective Action Guide exposure levels offsite for more than the immediate site area.

H

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

I

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

INDICATE: To point out or point to; to display the value of a process variable; to be a sign or symbol.

INITIATE: The act of placing equipment or a system into service, either manually or automatically. Activation of a function or protective feature (i.e. initiate a manual trip).

INJECTION: The act of forcing a fluid into a volume or vessel.

INOPERABLE: Not able to perform its intended function.

INTRUSION / INTRUDER is a person(s) present in a specified area without authorization. Discovery of a BOMB in a specified area is indication of INTRUSION into that area by an ADVERSARY.

L

LOSS: Failure of operability or lack of access to.

LOWER: To become progressively less in size, amount, number, or intensity.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

M

MAINTAIN: Take action, as necessary, to keep the value of the specified parameter within the applicable limits.

MONITOR: Observe and evaluate at a frequency sufficient to remain apprised of the value, trend, and rate of change of the specified parameter.

N

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONs.

NOTIFY: To give notice of or report the occurrence of; to make known to, to inform specified personnel; to advise; to communicate; to contact; to relay.

O

OPEN: To position a valve or damper so as to allow flow of the process fluid. To break an electrical connection which removes a power supply from an electrical device. To make available for entry or passage by turning back, removing, or clearing away.

OPERABLE: Able to perform its intended function.

P

PERFORM: To carry out an action; to accomplish; to affect; to reach an objective.

PRIMARY SYSTEM: The pipes, valves, and other equipment which connect directly to the Reactor Vessel or reactor coolant system such that a reduction in Reactor Vessel pressure will effect a lowering in the steam or water being discharged through an unisolated break in the system.

PROTECTED AREA boundary is within the security isolation zone and is defined in the PBNP Safeguards Contingency Plan.

R

REMOVE: To change the location or position of.

REPORT: To describe as being in a specific state.

REQUIRE: To demand as necessary or essential.

RESTORE: Take the appropriate action required to return the value of an identified parameter to within the acceptable limits.

RISE: Describes an increase in a parameter as the result of an operator or automatic action. To become progressively greater in size, amount, number or intensity.

RUPTURED: In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

S

SABOTAGE is deliberate damage, misalignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or

damaged due to malicious mischief may NOT meet the definition of SABOTAGE until this determination is made by security supervision.

SAFE PLANT SHUTDOWN: Hot or cold shutdown (reactor subcritical) with control of coolant inventory and decay heat removal.

SAFE SHUTDOWN SYSTEM: All cables, components, panels, power supplies, etc., necessary for a system to perform a safe shutdown function. Safe shutdown functions include: reactivity control, reactor coolant makeup, reactor heat removal, process system monitoring for variables necessary to control these functions and supporting functions such as component cooling, lubrication, etc., necessary for the operation of safe shutdown equipment.

SAMPLE: To perform an analysis on a specified media to determine its properties.

SHUTDOWN: To perform operations necessary to cause equipment to cease or suspend operation; to stop. "Shutdown unnecessary equipment."

SIGNIFICANT TRANSIENT is an UNPLANNED event involving one or more of the following: (1) automatic turbine runback >25% thermal reactor power, (2) electrical load rejection >25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations >10%.

SITE AREA EMERGENCY: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for 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.

SITE BOUNDARY: Per dose assessment methodology, the site boundary is approximately a one-mile radius around the site Protected Area.

STRIKE ACTION is a work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on PBNP. The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONs.

T

TRIP: To de-energize a pump or fan motor; to position a breaker so as to interrupt or prevent the flow of current in the associated circuit; to manually activate a semi-automatic feature.

U

UNAVAILABLE: Not able to perform its intended function.

UNCONTROLLED: An evolution lacking control but is not the result of an operator action.

UNPLANNED: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

UNUSUAL EVENT: Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

V

VALID: An indication, report, or condition is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator operability, the condition existence, or the report accuracy is removed. Implicit in this definition is the need for timely assessment.

VENT: To open an effluent (exhaust) flowpath from an enclosed volume; to reduce pressure in an enclosed volume.

VERIFY: To confirm a condition and take action to establish that condition if required. "Verify reactor trip."

VISIBLE DAMAGE is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREA is any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

Table R-0

Recognition Category R

Abnormal Rad Levels / Radiological Effluent

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
RU1 Any UNPLANNED Release of Gaseous or Liquid Radio-activity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. *Op. Modes: All*	**RA1** Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer. *Op. Modes: All*	**RS1** Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mrem TEDE or 500 mrem Thyroid CDE for the Actual or Projected Duration of the Release. *Op. Modes: All*	**RG1** Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. *Op. Modes: All*
RU2 Unexpected Rise in Plant Radiation. *Op. Modes: All*	**RA3** Release of Radioactive Material or Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown *Op. Modes: All*		
	RA2 Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. *Op. Modes: All*		

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ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU1

Initiating Condition – UNUSUAL EVENT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer.

Operating Mode Applicability: All

Emergency Action Levels: (RU1.1 or RU1.2 or RU1.3)

RU1.1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes or longer:

Table R-1 Radiation Monitors

Monitor	Reading
RE-214 Auxiliary Building Vent Exhaust Gas	2.04E-04 µCi/cc
RE-315 Auxiliary Building Exhaust Low Range Gas	2.04E-04 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-04 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-04 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-03 µCi/cc*
2RE-309 Containment Purge Exhaust High Range Gas	4.18E-03 µCi/cc*
RE-221 Drumming Area Ventilation Gas	3.16E-04 µCi/cc
RE-325 Drumming Area Exhaust Low Range Gas	3.16E-04 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-04 µCi/cc
1(2)RE-229 Service Water Overboard	5.56E-05 µCi/cc
RE-230 Waste Water Effluent	2.06E-03 µCi/cc**

* with Unit 2 Containment purge or forced vent not occurring
** with Waste Water Effluent discharge not isolated

RU1.3. Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times RETS.

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. PBNP incorporates features intended to control the release of radioactive effluents to the environment.

Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Radiological Effluent Technical Specifications (RETS) and implemented as described in the PBNP Offsite Dose Calculation Manual (ODCM) [Ref. 1, 2]. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The RETS multiples are specified in ICs RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate. Releases should not be prorated or averaged. For example, a release exceeding 4x RETS for 30 minutes does not meet the threshold for this IC.

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

RU1.1 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the alarm setpoint established by the radioactivity discharge permit and releases are not terminated within 60 minutes. The appropriate detailed response to an effluent alarm is described in the PBNP RMS Alarm Set Point and Response Book (RMSARB) [Ref.3]. These alarm setpoints may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the RETS. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

RU1.2 is intended for effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. All of the monitors included in Table R-1 provide monitoring on non-routine effluent release pathways for which a discharge permit would not normally be prepared. The reading used for the classification threshold is two times the ODCM default set point for the applicable radiation monitor.

RE-214, -315, -317, and -319 are noble gas monitors used to monitor all gaseous effluent releases occurring through the Primary Auxiliary Building ventilation stack (common for Unit 1 and 2).

2RE-307 and -309 are noble gas monitors used to monitor all gaseous effluent releases occurring from the Unit 1 and 2 Letdown Gas Strippers which discharge through the Unit 2 Containment Purge ventilation stack. 2RE-307 and 309 were selected because the detectors have a higher operating range than the Letdown Gas Stripper Building discharge monitor (RE-224), which enables them to provide indication for the classification threshold. The reading used for the classification threshold is two times the ODCM default set point for RE-224. This reading assumes that a planned batch release from the Unit 2 Containment building (purge or forced ventilation release) is not occurring. Batch releases from the Unit 2 containment building are controlled using a discharge permit.

RE-221, -325, and -327 are noble gas monitors used to monitor all gaseous effluent releases occurring from the Drumming Area ventilation stack (common for Unit 1 and 2).

1(2) RE-229 is a liquid monitor used to monitor all liquid effluent releases occurring from Unit 1 and 2 which discharge through the Service Water system.

RE-230 is a liquid monitor used to monitor all liquid effluent releases occurring from Unit 1 and 2 which discharge through the Wastewater Effluent system.

RU1.3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in lake water systems, etc.

PBNP Basis Reference(s):

1. RETS – PBNP Technical Specification 5.5.1 and 5.5.4
2. Offsite Dose Calculation Manual (ODCM)3. RMS Alarm Setpoint and Response Book (RMSASRB)
4. STPT 13.4, Radiation Monitoring System: Effluent Monitors

Initiating Condition – UNUSUAL EVENT

Unexpected Rise in Plant Radiation.

Operating Mode Applicability: All

Emergency Action Levels: (RU2.1 or RU2.2)

RU2.1. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water as indicated by any of the following:

- Spent fuel pool low water level alarm setpoint
- Visual observation

AND

Any UNPLANNED VALID Area Radiation Monitor reading rises as indicated by:

- 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

RU2.2. Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels
*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Basis:

This IC addresses increased radiation levels as a result of water level decreases above the Reactor Vessel flange or events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and may represent a potential degradation in the level of safety of the plant [Ref. 1].

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via RU2.1 is appropriate given their potential for increased doses to plant staff. Classification as an UE is warranted as a precursor to a more serious event. Indications include instrumentation such as water level and local area radiation monitors [Ref. 2], and personnel (e.g., refueling crew) reports. If available, security video cameras may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or decrease in refueling water storage tank level.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the reading

on an area radiation monitor located near the refueling bridge may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Generally, increased radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss. For refueling events where the water level drops below the Reactor Vessel flange classification would be via CU2. This event escalates to an Alert per IC RA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Matrix for events in operating modes 1-4.

The low level alarm is actuated by LC-634, SFP Level Indicator at 62'-8" based on maintaining at least 6' of water on a withdrawn fuel assembly [Ref. 2].

RU2.2 addresses UNPLANNED rises in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. This event escalates to an Alert per IC RA3 if the increase in dose rates impedes personnel access necessary for safe operation.

PBNP Basis Reference(s):

1. RMS Alarm Response Book (RMSARB)
2. DBD-13 Spent Fuel Pool Cooling and Filtration
3. STPT 13.1 Area Monitors

Initiating Condition – ALERT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer.

Operating Mode Applicability: All

Emergency Action Levels: (RA1.1 or RA1.2 or RA1.3)

RA1.1. VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

RA1.2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes or longer:

Table R-2 Radiation Monitors

Monitor	Reading
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-02 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-02 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-01 µCi/cc*
2RE-309 Containment Purge Exhaust High Range	4.18E-01 µCi/cc*
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-02 µCi/cc

* with Unit 2 Containment purge or forced vent not occurring

RA1.3. Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times RETS.

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. PBNP incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Radiological Effluent Technical Specifications (RETS) and implemented as described in the PBNP Offsite Dose Calculation Manual (ODCM) [Ref. 1, 2]. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The RETS multiples are specified in ICs RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate [Ref. 2], the emphasis in classifying these events is the degradation in the level of safety

of the plant, NOT the magnitude of the associated dose or dose rate. Releases should not be prorated or averaged.

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

RA1.1 addresses radioactivity releases that for whatever reason cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. The appropriate detailed response to an effluent alarm is described in the PBNP RMS Alarm Set Point and Response Book (RMSASRB) [Ref. 3]. These alarm setpoints may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the RETS. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

RA1.2 addresses effluent or accident radiation monitors on non-routine release pathways (i.e., for which a discharge permit would not normally be prepared). The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. All of the monitors included in Table R-2 provide monitoring on non-routine effluent release pathways for which a discharge permit would not normally be prepared. The reading used for the classification threshold is two hundred times the ODCM default set point for the applicable radiation monitor.

RE-317 and -319 are noble gas monitors used to monitor all gaseous effluent releases occurring through the Primary Auxiliary Building (PAB) ventilation stack (common for Unit 1 and 2). RE-317 and -319 were selected because the detectors have a higher operating range than the PAB ventilation stack radiation monitor (RE-214), which enables them to provide indication for the classification threshold.

2RE-307 and -309 are noble gas monitors used to monitor all gaseous effluent releases occurring from the Unit 1 and 2 Letdown Gas Strippers which discharge through the Unit 2 Containment Purge ventilation stack. 2RE-307 and -309 were selected because the detectors have a higher operating range than the Letdown Gas Stripper Building discharge monitor (RE-224), which enables them to provide indication for the classification threshold. The reading used for the classification threshold is two hundred times the ODCM default set point for RE-224. This reading assumes that a planned batch release from the Unit 2 Containment building (purge or forced ventilation release) is not occurring. Batch releases from the Unit 2 containment building are controlled using a discharge permit.

RE-327 is a noble gas monitor used to monitor all gaseous effluent releases occurring from the Drumming Area ventilation stack (common for Unit 1 and 2). RE-327 was selected because the detector has a higher operating range than the Drumming Area ventilation stack radiation monitor (RE-221), which enables it to provide indication for the classification threshold. The reading used for the classification threshold is two hundred times the ODCM default set point for RE-221.

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 through RA1.1 or RA1.3 (e.g., sampling initiated due to entry into EAL RU1).

RA1.3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in lake water systems, etc.

Due to the uncertainty associated with meteorology, emergency implementing procedures call for the timely performance of dose assessments using actual (real-time) meteorology in the event of a gaseous radioactivity release of this magnitude. The results of these assessments should be compared to the ICs RS1 agbnd RG1 to determine if the event classification should be escalated.

PBNP Basis Reference(s):

1. RETS – PBNP Technical Specifications 5.5.1 and 5.5.4
2. Offsite Dose Calculation Manual (ODCM)
3. RMS Alarm Setpoint and Response Book (RMSASRB)
4. STPT 13.4, Radiation Monitoring System: Effluent Monitors

Initiating Condition – ALERT

Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

Operating Mode Applicability: All

Emergency Action Levels: (RA2.1 or RA2.2)

RA2.1. A VALID high alarm or reading on any of the following radiation monitors:

- o RE-105 SFP Area Low Range Radiation Monitor
- o RE-135 SFP Area High Range Radiation Monitor
- o RE 221 Drumming Area Ventilation Gas Monitor
- o RE 321 Drumming Area Exhaust Beta Particulate Monitor
- o RE 325 Drumming Area Exhaust Low Range Gas Monitor
- o 1(2) RE 102 El. 66' Containment Low Range Monitor
- o 1(2) RE-211 Containment Air Particulate Monitor
- o 1(2) RE 212 Containment Noble Gas Monitor

RA2.2. Water level LESS THAN 10 ft above an irradiated fuel assembly for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering.

Basis:

This IC addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent a degradation in the level of safety of the plant. These events escalate from IC RU2 in that fuel activity has been released, or is anticipated due to fuel heatup. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in IC EU1.

RA2.1 addresses radiation monitor indications [Ref. 1, 2, 3] of fuel uncovery and/or fuel damage. Increased readings on ventilation monitors may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Application of these Initiating Conditions requires understanding of the actual radiological conditions present in the vicinity of the monitor. VALID high alarms indicated by the radiation monitors listed in RA2.1 may be indicative of a fuel handling accident and are, therefore, appropriate for this EAL [Ref. 1, 2, 3]. High alarm setpoint values and the appropriate detailed

responses to radiation monitor high alarms are provided in the PBNP RMS Alarm Set Point and Response Book (RMSASRB) [Ref.1] Information Notice No. 90-08, *"KR-85 Hazards from Decayed Fuel"* was considered in establishing radiation monitor EAL thresholds and there is no impact on this EAL.

Entry into RA2.2 is based on personnel (e.g., refueling crew) reports. There is no site-specific water level instrumentation available that could be used for entry into this EAL. Water level lowering to less than 10 feet above an irradiated fuel assembly is indicative of conditions that will result in irradiated fuel uncovering while maintaining adequate radiation shielding to protect personnel in the area [Ref.7]. This EAL does not apply to planned activities that might require an irradiated fuel assembly to be raised to a level that is less than 10 feet from the surface of the water (e.g., maintenance or repair).

Escalation, if appropriate, would occur via IC RS1 or RG1 or Emergency Director judgment.

PBNP Basis Reference(s):

1. RMS Alarm Setpoint and Response Book (RMSASRB)
2. AOP-8B Irradiated Fuel Handling Accident in Containment
3. AOP-8C Fuel Handling Accident in PAB
4. STPT 13.1 Area Monitors
5. STPT 13.2 Process Monitors

6STP 13.4 Radiation Monitoring System: Effluent Monitors

7. DBD-05, Fuel Handling System

Initiating Condition -- ALERT

Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown

Operating Mode Applicability: All

Emergency Action Levels: (RA3.1 or RA3.2)

RA3.1. VALID radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:

Control Room (RE-101)

OR

Central Alarm Station (by survey)

OR

Secondary Alarm Station (by survey)

RA3.2. Any VALID radiation monitor reading GREATER THAN 1 R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1).

Table H-1 Vital Areas
• 1(2) Containment Building • Primary Auxiliary Building • Turbine Building (by survey) • Control Building • Diesel Generator Building (by survey) • Gas Turbine Building (by survey) • Circ Water Pump House (by survey)

Basis:

This IC addresses increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this IC. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. For example, a dose rate of 15 mR/hr in the control room may be a problem in itself. However, the increase may also be indicative

of high dose rates in the containment due to a LOCA. In this latter case, an SAE or GE may be indicated by the fission product barrier matrix ICs.

At PBNP, the example EALs could result in declaration of an Alert at one unit due to a radioactivity release or radiation shine resulting from a major accident at the other unit. This is appropriate if the increase impairs operations at the operating unit.
This IC is not meant to apply to increases in the containment high range radiation monitors which normally indicate 1.0-2.5 R/hr, as these are events which are addressed in the fission product barrier matrix ICs. Nor is it intended to apply to anticipated temporary increases due to planned events (e.g., radwaste container movement, incore detector movement, radiography, movement of large components, depleted resin transfers, etc.)

For RA3.1 areas requiring continuous occupancy include the Control Room, the central alarm station (CAS) and the secondary alarm station (SAS). The CAS and SAS have no installed radiation monitoring capability [Ref. 3], therefore entry into this EAL is based on radiation surveys in these areas. 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. [Ref. 2, 3]

For RA3.2 areas requiring infrequent access, a valid radiation monitor reading greater than 1 R/hr would result in additional exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (e.g., current radiation protection and ALARA procedures, NMC administrative exposure limits, 10CFR20 limits, etc) and would impede necessary access.

As used here, *impede,* includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant.

The Turbine Building, Diesel Generator Building, Gas Turbine Building and Circ Water Pump House have no installed radiation monitor capability [Ref. 3], therefore entry into the EAL is based on radiation surveys in these areas.

Areas listed in Table H-1 were selected because they are areas or contiguous to areas requiring access to maintain plant safety functions.

PBNP Basis Reference(s):

1. GDC 19
2. NUREG-0737, "Clarification of TMI Action Plan Requirements", Section III.D.3
3. RMS Alarm Setpoint and Response Book (RMSASRB)

Initiating Condition -- SITE AREA EMERGENCY

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mrem TEDE or 500 mrem Thyroid CDE for the Actual or Projected Duration of the Release.

Operating Mode Applicability: All

Emergency Action Levels: (RS1.1 or RS1.2 or RS1.3)

Note: *If dose assessment results are available at the time of declaration, the classification should be based on RS1.2 instead of RS1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

RS1.1. VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Table R-3 Radiation Monitors	
Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+00 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+00 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E-01 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E-01 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E-01 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-02 µCi/cc
1 S/G Safety Valve open	2.48E-02 µCi/cc
2 S/G Safety Valves open	1.24E-02 µCi/cc
3 S/G Safety Valves open	8.25E-03 µCi/cc
4 S/G Safety Valves open	6.20E-03 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 closed window dose rates exceeding 100 mrem/hr expected to continue for more than one hour, at or beyond the site boundary;

OR

Analysis of field survey samples indicate thyroid CDE of 500 mrem for one hour of inhalation, at or beyond the site boundary.

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed a small fraction of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone, e.g., fuel handling accident in spent fuel building.

The 100 mrem TEDE dose is set at 10% of the EPA PAG, while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The Table R-3 monitor list in RS1.1 includes monitors on all potential gaseous effluent release pathways. These monitors were selected because the detectors have an operating range that enables them to provide indication for the classification threshold.

1(2) RE-307 and –309 are noble gas monitors used to monitor all releases occurring from the Unit and 2 Containment purge ventilation stacks. RE-317 and –319 are noble gas monitors used to monitor all releases occurring from the Drumming Area ventilation stack (common for Unit 1 and 2). 1(2)RE-231 and 1(2)RE-232 are process monitors used to monitor all releases occurring through the Unit 1 and 2 atmospheric steam dump and S/G safety valves.

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.
The monitor reading thresholds for RS1.1 were determined by multiplying the monitor readings in EAL RG1.1 Table R-4 by 0.1 to determine the monitor reading thresholds corresponding to 500 mrem Thyroid CDE at or beyond the site boundary (1 mile downwind), which is the more limiting dose value for these calculations [Ref.1]. The inputs used for the calculations in Reference 1 are described in the Basis section for EAL RG1.

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

PBNP Basis Reference(s):

1. Effluent Monitor Classification Thresholds WEDAP Basis Document, NPC 2004-00731
2. EPIP 1.3 Dose Assessment and Protective Action Recommendations
3. FSAR Table 2.6-3 Stability Index Distribution
4. FSAR Table 2.6-4 Site Atmospheric Stability Analysis Annual Average 13 month Data
5. FSAR Figure 2.6-2 Stability Class Distribution in Percent of Total Observed
6. USEPA 400-R-92-001, Manual of Protective Action Guidelines and Protective Actions for Nuclear Incidents

7. DBD-T-46 Section 3.1, Station Blackout

Initiating Condition – GENERAL EMERGENCY

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability: All

Emergency Action Levels: (RG1.1 or RG1.2 or RG1.3)

Note: *If dose assessment results are available at the time of declaration, the classification should be based on RG1.2 instead of RG1.1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

RG1.1. VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Table R-4 Radiation Monitors	
Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+01 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+01 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E+00 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E+00 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E+00 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-01 µCi/cc
1 S/G Safety Valve open	2.48E-01 µCi/cc
2 S/G Safety Valves open	1.24E-01 µCi/cc
3 S/G Safety Valves open	8.25E-02 µCi/cc
4 S/G Safety Valves open	6.20E-02 µCi/cc

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 closed window dose rates exceeding 1000 mrem/hr expected to continue for more than one hour, at or beyond site boundary.

OR

Analysis of field survey samples indicate thyroid CDE of 5000 mrem for one hour of inhalation, at or beyond site boundary.

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the

protection of the public and likely involve fuel damage. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The Table R-4 monitor list in RG1.1 includes monitors on all potential gaseous effluent release pathways. These monitors were selected because the detectors have a high enough operating range that enables them to provide indication for the classification threshold.

1(2) RE-307 and –309 are noble gas monitors used to monitor all releases occurring from the Unit 1 and 2 Containment purge ventilation stacks. RE-317 and –319 are noble gas monitors used to monitor all releases occurring through Primary Auxiliary Building (PAB) ventilation stack 9common for Unit 1 and 2). RE-327 is a noble gas monitor used to monitor all releases occurring from the Drumming Area ventilation stack (common for Unit 1 and 2). 1(2)RE-231 and 1(2)RE-232 are process monitors used to monitor all releases occurring through the Unit 1 and 2 atmospheric steam dump and S/G safety valves.

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.

The monitor reading thresholds for RG1.1 were determined using the Wisconsin Electric Dose Assessment Program (WEDAP) computer code and annual average meteorology [Ref.1]. The monitor readings determined in Reference 1 were back calculated from a dose value of 5000 mrem Thyroid CDE at or beyond the site boundary (1 mile downwind), which is the more limiting dose value for these calculations. The inputs used for the calculations in Reference 1 are as follows:

Wind direction: from 30 degrees (NNE to SSW)
Basis: FSAR Table 2.6-4 Annual average meteorology
Wind speed: 10 mph
Basis: FSAR Table 2.6-4 Annual average meteorology for stability class D (8-12 mph)
Stability class: D
Basis: FSAR Tables 2.6-2 and 2.6-3 Annual average meteorology - Stability index distribution and Stability Class Distribution in Percent of Total Observed and WEDAP default stability class
Time after shutdown: 4 hrs.
Basis: FSAR Appendix A-1 and DBD-T-46 Section 3.1
Station Blackout Coping Time (time to core damage)
Release duration: 4 hrs. (default)
Containment purge vs. forced vent: 1 purge fan
Releases filtered except for 1(2) RE 231/232 Steam Line Monitors
No lake breeze effect
No precipitation
Building Wake Effect (default)

Source term: LOCA/Gap release inside containment except for 1(2) RE 231/232 Steam line Monitors which were based on Gap release/SGTR
No containment sprays
SGTR release path assumes SG water level <29% narrow range
Calculation results for RE 231were used for monitor reading threshold for RE 232 (similar monitor and location)

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

PBNP Basis Reference(s):

1. Effluent Monitor Classification Thresholds WEDAP Basis Document, NPC 2004-00731
2. EPIP 1.3 Dose Assessment and Protective Action Recommendations
3. FSAR Table 2.6-3 Stability Index Distribution
4. FSAR Table 2.6-4 Site Atmospheric Stability Analysis Annual Average 13 month Data
5. FSAR Figure 2.6-2 Stability Class Distribution in Percent of Total Observed
6. UDSEPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
7. DBD-T-46 Section 3.1, Station Blackout
8. FSAR Appendix A-1, Station Blackout

Table C-0
Recognition Category C
Cold Shutdown/Refueling System Malfunction
INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
CU1 RCS Leakage. *Op. Mode: Cold Shutdown*	**CA1** Loss of RCS Inventory. *Op. Modes: Cold Shutdown*	**CS1** Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability. *Op. Modes: Cold Shutdown*	**CG1** Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel. Op. Modes: Cold Shutdown, Refueling
CU2 UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel *Op. Mode: Refueling*	**CA2** Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel. *Op. Modes: Refueling*	**CS2** Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel. Op. Modes: Refueling	
CU3 Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes. *Op. Modes: Cold Shutdown, Refueling*	**CA3** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Cold Shutdown, Refueling, Defueled*		
CU4 UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel. *OP. Modes: Cold Shutdown, Refueling*	**CA4** Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel. *Op. Modes: Cold Shutdown, Refueling*		
CU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities. *Op. Modes: Cold Shutdown, Refueling*			
CU7 UNPLANNED Loss of Required DC Power for Greater than 15 Minutes. *Op. Modes: Cold Shutdown, Refueling*			

CU8 Inadvertent Criticality.
Op Modes:, Cold Shutdown, Refueling

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Initiating Condition -- UNUSUAL EVENT

RCS Leakage.

Operating Mode Applicability: Cold Shutdown

Emergency Action Levels: (CU1.1 or CU1.2)

CU1.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm.

CU1.2. Identified leakage GREATER THAN 25 gpm.

Basis:

This IC is included as a UE because it is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is sufficiently large to be observable via normally installed instrumentation (e.g., Pressurizer level, RCS loop level instrumentation, etc...). Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). OI 55 provides instructions for calculating primary system leak rate by water inventory balances. The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Prolonged loss of RCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel).

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and RCS inventory and level monitoring means such as Pressurizer level indication and makeup volume control tank levels are normally available. In the refueling mode the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

PBNP Basis Reference(s):

1. TS 3.4.13, RCS Operational Leakage limits
2. OI 55, Primary Leak Rate Calculation
3. OM 3.19, Reactor Coolant System Leakage Determination

CU2

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Refueling

Emergency Action Levels: (CU2.1 or CU2.2)

CU2.1. UNPLANNED RCS level lowering below the Reactor Vessel flange (89.1%) for GREATER THAN OR EQUAL TO 15 minutes

CU2.2. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise

AND

Reactor Vessel level cannot be monitored

Basis:

This IC is included as an UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Refueling evolutions that decrease RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the Reactor Vessel flange warrants declaration of an Unusual Event due to the reduced RCS inventory that is available to keep the core covered. The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists. Continued loss of RCS Inventory will result in escalation to the Alert level via either IC CA2 (Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel).

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

In the refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be used (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing Containment Sump A and Waste Holdup Tank level changes [Ref. 1, 2]. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. OI 55 [Ref. 4] provides instructions for calculating primary system leak rate by water inventory balances. Containment Sump A is equipped with a high level alarm (80%). Escalation to Alert would be via either CA2 or RCS heatup via CA4.

CU2.1 involves a decrease in RCS level below the top of the Reactor Vessel flange that continues for 15 minutes due to an UNPLANNED event. The Reactor Vessel flange is at elevation 40 ft 8 in., which is 89% on LI-447/447A or 89 in. on LI-447B [Ref. 3]. This EAL is not applicable to decreases in flooded reactor cavity level (covered by RU2.1) until such time as the level decreases to the level of the vessel flange. If Reactor Vessel level continues to decrease and reaches the Bottom ID of the RCS Loop, (33 ft 2-7/8 in. elev. or 0%/0 in.), then escalation to CA2 would be appropriate. Note that the Bottom ID of the RCS Loop Setpoint corresponds to the bottom of the Reactor Vessel loop penetration (not the low point of the loop).

PBNP Basis Reference(s):

1. ARB C01 B 1-4, UNIT 1 CONTAINMENT SUMP A LEVEL HIGH
2. STPT 12.1, Waste Disposal System
3. OP 4D Part 3, Draining the Reactor Cavity and Reactor Coolant System, Table 1
4. OI 55, Primary Leak Rate Calculation

CU3

Initiating Condition – UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CU3.1. Loss of all **offsite** power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN 15 minutes.

AND

At least 1 emergency generator is supplying power to an emergency bus.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The 4160 VAC system includes two safety-related (essential) buses per unit, 1(2)-A05 (A train) and 1(2)-A06 (B train). Offsite power from the 345 KVAC system is stepped down through the 13.8 KVAC system to the Low Voltage Station Auxiliary Transformer (LVSAT) 1(2)-X04. The LVSATs provide power to 4160 VAC switching buses 1(2)-A03 and 1(2)-A04, which in turn provide power to safety-related buses 1(2)-A05 and 1(2)-A06.

During emergency or abnormal situations, the 4160 VAC system is supplied by emergency diesel generators (G01 through G04) or the gas turbine generator (G05). Following a loss of power, ECA 0.0 provides guidance to restore power to any 4160 VAC safety-related bus. For the purpose of classification under this EAL, offsite power sources include any of the following:

- 345 KVAC system supplying power to the 13.8 KVAC system and the unit LVSATs
- Cross-tying with the opposite unit power supply
- Backfeeding power to the UATs through the 19 KVAC system and the main step-up transformer X-01. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source.

PBNP has the capability to cross-tie AC power from the other unit and therefore takes credit for the redundant power source for this IC. However, the inability to effect the cross-tie within 15 minutes warrants declaring a UE.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
2. DBD-18, 13.8 KVAC System, Section 3.3.0
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A U1, Main Power Transformer Backfeed

Initiating Condition – UNUSUAL EVENT

UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (CU4.1 or CU4.2)

CU4.1. An UNPLANNED event results in RCS temperature exceeding 200°F

CU4.2. Loss of all RCS temperature and Reactor Vessel level indication for GREATER THAN 15 minutes.

Basis:

This IC is included as an UE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered. In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). In addition, the operators should be able to monitor RCS temperature and Reactor Vessel level so that escalation to the alert level via CA4 or CA1 will occur if required.

During refueling the level in the Reactor Vessel will normally be maintained above the Reactor Vessel flange. Refueling evolutions that decrease water level below the Reactor Vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/Reactor Vessel temperatures depending on the time since shutdown. Escalation to the Alert level is via CA4.

Unlike the cold shutdown mode, normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown or refueling modes, CU4.2 would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via CA2 based on an inventory loss or CA4 based on exceeding its temperature criteria (200°F) [Ref. 1].

Reactor Vessel water level is normally monitored using the following instruments [Ref. 2, 4]:

- LT-494, 495 (RVLIS) Reactor Vessel WR Water Level (0 - 125 ft)
- LT-496, 497 (RVLIS) Reactor Vessel NR Water Level (0 - 45 ft)
- LT-447, 447A Reduced Inventory RV Water Level (0 - 100 in.)

LT-494, 495, 496, 497 provide appropriate level signals to allow the Control Room operator to monitor the associated parameter during design basis accidents. These instruments are only to be used for trend information and are not used for definitive Reactor Vessel level indication when draining the reactor cavity or RCS.

LT-447, 447A provide Reactor Vessel water level indication in the Control Room during reduced RCS inventory condition. They are calibrated to indicate level from approximately 10 in. above the vessel flange to a level approximately 90 inches below the vessel flange (bottom of the hot leg). The instrument scale range is 0 -100% which corresponds to 0 -100 in. The readout can be displayed on the Plant Process Computer System (PPCS) at the operator's desk and on the control board with a readout of ±0.1 inches while at mid-loop. This provides the ability to read the level from the bottom of the reactor coolant pipe (0% or 33 ft 2-7/8 in. elev.) to well above the Reactor Vessel head flange (89.1% or 40 ft 8 in. elev.).

Local indicator LI-447B, RC Reduced Inventory Level Indicator (0 -100 in.), also provides Reactor Vessel water level indication. 0 in. on LI-447B corresponds to 0% on LI-447/447A.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include [Ref. 2, 3]:

- Tavg instrumentation
- TE-1 thru 39, Core Exit Thermocouples (50 - 1600°F)
- TE-450A, 450C, 451A, 451C, Cold Leg Loop WR Temperature (50 - 750°F)
- TE-450B, 450D, 451B, 451D, Hot Leg Loop WR Temperature (50 - 750°F)
- TE-630, RHR Inlet Temperature

Loop WR Temperature Elements provide wide range RCS temperature signals for monitoring heatup and cooldown, and unusual events such as natural circulation, where the RTDs in the loop bypass lines do not provide meaningful temperature signals. Only Loop B Cold Leg WR temperature indication uncertainties have been analyzed to ensure the PTLR limits are maintained. Loop B cold leg WR temperature channel is used when on RHR. TC_{AVG} is preferred thermocouple indication and provides an average of all thirty-nine thermocouples and the trend recorders can look at one of eight or an average of all eight thermocouples.

The Emergency Director must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

PBNP Basis Reference(s):

1. Technical Specifications Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206

2. DBD-09, Reactor Coolant System, Rev. 6, Sections 3.23.1 to 3.23.4, 3.24.1, 3.24.2, 3.25.1 and 3.25.2
3. OI 105, RCS Heatup/Cooldown Plotting, Rev. 9
4. OP 4D Part 3, Draining the Reactor Cavity and Reactor Coolant System, Table 1, Rev. 16

SYSTEM MALFUNCTION

CU6

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (CU6.1 or CU6.2)

CU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Plant Public Address System • Security Radio • Commercial Phone System • Portable radios • Sound power phones

CU6.2. Loss of all Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• Emergency Notification System (ENS) • Health Physics Network (HPN) • Operations Control Counterpart Link (OCCL) • Management Counterpart Link (MCL) • Protective Measures Counterpart Link (PMCL) • Reactor Safety Counterpart Link (RSCL) • Nuclear Accident Reporting System (NARS) • Commercial Phone System • General Telephone Lines • Manitowoc County Sheriff's Department Radio

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

Table C-1 onsite communications loss [Ref. 1, 2] encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system and radios / walkie talkies).

Table C-2 offsite communications loss [Ref. 1, 2] encompasses the loss of all means of communications with offsite authorities. This includes the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

PBNP Basis Reference(s):

1. EPMP 2.1, Testing of Communications Equipment
2. EPMP 2.1A, Monthly Communications Test

SYSTEM MALFUNCTION

CU7

Initiating Condition – UNUSUAL EVENT

UNPLANNED Loss of Required DC Power for GREATER THAN 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CU7.1 UNPLANNED Loss of vital DC power to required DC busses based on bus voltage indications LESS THAN 115 VDC

AND

Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

UNPLANNED is included in this IC and EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely plants will perform maintenance on a Train related basis during shutdown periods. It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per CA4 "Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel."

LESS THAN 115 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment [Ref. 1, 2].

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

The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses.

PBNP Basis Reference(s):

1. FSAR Section 8.7
2. 0-SOP-DC-001/2/3/4 125 VDC System, Bus D-01, D-02, D-03, D-04 and Components

CU8

Initiating Condition – UNUSUAL EVENT

Inadvertent Criticality.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CU8.1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events and inadvertent dilution events. This IC indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated) which are addressed in the companion IC SU8.

This condition can be identified using the startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned fuel bundle or control rod movements during core alterations. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

This condition can be identified using startup rate monitors (NI-31D/32D - Source Range Startup Rate, and NI-35D/36D - Intermediate Range Startup Rate) [Ref. 1].

Escalation would be by Emergency Director Judgment.

PBNP Basis Reference(s):

1. OP 1B, Reactor Startup, Rev 50,Step 5.1 and 5.17.15

Initiating Condition – ALERT

Loss of RCS Inventory.

Operating Mode Applicability: Cold Shutdown

Emergency Action Levels: (CA1.1 or CA1.2)

CA1.1. Loss of RCS inventory as indicated by Reactor Vessel level LESS THAN 6% on LI-447 and LI-447A

CA1.2. Loss of RCS inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise

AND

RCS level cannot be monitored for GREATER THAN 15 minutes

Basis:

These EALs serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel level decrease and potential core uncovery. The LI-447 and 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. The 6 inch level of the RCS Loop was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred [Ref. 1]. The 6 inch level is above the CS2 setpoint of 0, the level equal to the bottom of the Reactor Vessel loop penetration. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the cold shutdown mode, normal RCS level and Reactor Vessel level instrumentation systems will normally be available. During preparations for refueling, reactor vessel level indication may not be available. In this instance, classification would be completed under CA1.2 for sump and tank level indications. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure

they are indicative of RCS leakage [Ref. 1, 2]. The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency EAL duration. The 15-minute duration allows CA1 to be an effective precursor to CS1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS1 basis. Therefore this EAL meets the definition for an Alert emergency.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

If Reactor Vessel level continues to decrease then escalation to Site Area will be via CS1 (Loss of Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel).

PBNP Basis Reference(s):

1. OP 4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
2. OM 3.19, REACTOR COOLANT SYSTEM LEAKAGE DETERMINATION

Initiating Condition -- ALERT

Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Refueling

Emergency Action Levels: (CA2.1 or CA2.2)

CA2.1. Loss of Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 6% on LI-447 / LI-447A .

CA2.2. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise

AND

Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes

Basis:

These EALs serve as precursors to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel level decrease and potential core uncovery. The LI-447 / LI-447A threshold corresponds to the 6 inches above the bottom inside diameter of the RCS loop [Ref. 1]. The 6 inch level of the RCS Loop was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. This condition will result in a minimum classification of Alert. The 6 inch level is above the CS2 setpoint of 0, the level equal to the bottom of the Reactor Vessel loop penetration. The Bottom ID of the RCS Loop Setpoint is the level equal to the bottom of the Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally used (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes [Ref. 1, 2]. Sump and tank level rises must be evaluated against other potential sources

of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS2 Site Area Emergency EAL duration. The 15-minute duration allows CA2 to be an effective precursor to CS2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS2 basis. Therefore this EAL meets the definition for an Alert.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

If Reactor Vessel level continues to decrease then escalation to Site Area will be via CS1 (Loss of Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel).

PBNP Basis Reference(s):

1. OP 4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
2. OM 3.19, REACTOR COOLANT SYSTEM LEAKAGE DETERMINATION

SYSTEM MALFUNCTION

CA3

Initiating Condition – ALERT

Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability: Cold Shutdown
Refueling
Defueled

Emergency Action Level:

CA3.1. Loss of all offsite power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06

AND

Failure of all emergency generators to supply power to emergency busses.

AND

Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Escalating to Site Area Emergency IC SS1, if appropriate, is by Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

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 essential busses. Even though an essential 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.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
2. DBD-18, 13.8 KVAC System, Section 3.3.0
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A U1 Main Power Transformer Backfeed

SYSTEM MALFUNCTION

CA4

Initiating Condition – ALERT

Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (EAL CA4.1 or CA4.2 or CA4.3)

CA4.1. With CONTAINMENT CLOSURE and RCS integrity not established an UNPLANNED event results in RCS temperature exceeding 200 degrees F.

CA4.2. With CONTAINMENT CLOSURE established and RCS integrity not established or RCS inventory reduced an UNPLANNED event results in RCS temperature exceeding 200 degrees F for GREATER THAN 20 minutes[1].

CA4.3. An UNPLANNED event results in RCS temperature exceeding 200 degrees F for GREATER THAN 60 minutes[1] or results in an RCS pressure rise of GREATER THAN 10 psig.

Basis:

CA4.1 addresses complete loss of functions required for core cooling during refueling and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). No delay time is allowed for CA4.1 because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

CA4.2 addresses the complete loss of functions required for core cooling for GREATER THAN 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is reduced (e.g., mid loop operation). As in CA4.1, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed later in this basis) and is believed to be conservative given that a low pressure Containment barrier to fission product release is established. Note 1 indicates that CA4.2 is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 20 minute time frame.

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[1]Note: if an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.

CA4.3 addresses complete loss of functions required for core cooling for GREATER THAN 60 minutes during refueling and cold shutdown modes when RCS integrity is established. As in CA4.1 and CA4.2, RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this EAL is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The 10 psig pressure rise covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. Pressure indicators PT-420 (RCS loop pressure) and PT-493 (pressurizer pressure) have a range of 0 to 1000 psig and are capable of measuring pressure to less than 10 psig [Ref. 6]. Note 1 indicates that CA4.3 is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 60 minute time frame assuming that the RCS pressure rise has remained less than the site specific pressure value. EAL CA4.3 does not apply when Pressurizer is solid. Minor temperature changes during solid plant operations will cause dramatic pressure swings. The described conditions, "RCS integrity not established" sets the "Pressurizer not solid" initial condition for this EAL, therefore this EAL does not apply when Pressurizer is solid.

Escalation to Site Area would be via CS1 or CS2 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

This IC and its associated EALs are based on 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 sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above 200 degreesF when the heat removal function is available.

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

PBNP Basis Reference(s):

1. Technical Specifications Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
2. OP 1A, Cold Shutdown to Hot Standby
3. Technical Specifications B 3.6.1, Containment, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
4. CL 1E, Containment Closure Checklist
5. OP 4F, Reactor Coolant System Reduced Inventory Requirements
6. DBD-09, Reactor Coolant System

Initiating Condition – SITE AREA EMERGENCY

Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability.

Operating Mode Applicability: Cold Shutdown

Emergency Action Levels: (CS1.1 or CS1.2)

CS1.1. With CONTAINMENT CLOSURE not established:

a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indication on LI-447/LI-447A

OR

b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise

CS1.2. With CONTAINMENT CLOSURE established:

a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indication on LI-447/LI-447A

OR

b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either:
   - Unexplained Containment Sump A or Waste Holdup Tank level rise
   - Erratic Source Range Monitor Indication

Basis:

Under the conditions specified by this IC, continued decrease in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to an Reactor Vessel breach, pressure boundary leakage, or continued boiling in the Reactor Vessel.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

In the cold shutdown mode, normal RCS level and reactor vessel level indication systems (RVLIS) will normally be available. During preparations for refueling, reactor vessel level indication may not be available. In this instance, classification would be completed under CS1.2 for sump and tank level indications. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

These EALs are based on 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.* A number of variables, (PWRs - e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30-minutes was chosen.

The level associated with or without CONTAINMENT CLOSURE corresponds to the bottom inside diameter of the RCS loop.

The 30-minute duration allowed when CONTAINMENT CLOSURE is established allows sufficient time for actions to be performed to recover needed cooling equipment and is considered to be conservative given that level is being monitored via CS1 and CS2. Effluent release is not expected with closure established.
Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel) or radiological effluent IC RG1 (Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

PBNP Basis Reference(s):

1. PBNP FSAR 4.0 Reactor Coolant System Design Basis
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. OI 55, Primary Leak Rate Calculation

SYSTEM MALFUNCTION

CS2

Initiating Condition – SITE AREA EMERGENCY

Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Refueling

Emergency Action Levels: (CS2.1 or CS2.2)

CS2.1. With CONTAINMENT CLOSURE not established:

a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indicated on LI-447/LI-447A

OR

b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following:
   - Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr
   - Erratic Source Range Monitor Indication

CS2.2. With CONTAINMENT CLOSURE established

a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indicated on LI-447/LI-447A

OR

b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by one or more of the following:
   - Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr
   - Erratic Source Range Monitor Indication

Basis:

Under the conditions specified by this IC, continued decrease in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to a Reactor Vessel breach or continued boiling in the Reactor Vessel.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is

completed. Entry into the refueling mode procedurally may not occur for typically 100 or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

These EALs are based on 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.* A number of variables, (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier.

The level associated with or without CONTAINMENT CLOSURE corresponds to the bottom inside diameter of the RCS loop. The level associated with containment closure established corresponds to the top of active fuel.

In Refueling mode, normal RCS level indication (e.g., RVLIS) may be unavailable but alternate means of level indication are normally available to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment High Range Monitor indication and possible alarm. Typical readings at full power are 1.0 to 2.5 R/hr. The Containment High Radiation indicators (1(2)RM126, 127 and 128) are log scaled with a span of $1\text{-}10^8$ R/hr. The 10 R/hr reading has been selected to be well above that expected under normal plant conditions [Ref. 1].

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (NIS N-31 and N-32) can be used as a tool for making such determinations.

For CS2.2 in the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally available (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

Effluent release is not expected with closure established. Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel) or radiological effluent IC AG1 (Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

PBNP Basis Reference(s):

1. Eng Eval 2001-28, Containment High Radiation Channel Check Tolerance, 10/5/01
2. PBNP FSAR 4.0 Reactor Coolant System Design Basis, Table 4.1-6

Initiating Condition -- GENERAL EMERGENCY

Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CG1.1. Reactor Vessel Level:

1. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Hold Up Tank level rise or any other indication of loss of Reactor Vessel inventory

   AND

2. Reactor Vessel level:
   a. LESS THAN [30 ft] 27 ft (RVLIS NR) or 0% indicated on LI-447/LI-447A for GREATER THAN 30 minutes

      OR

   b. cannot be monitored with indication of core uncovery for GREATER THAN 30 minutes as evidenced by any of the following:
      - Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr
      - Erratic Source Range Monitor Indication

      AND

3. CONTAINMENT challenged as indicated by any of the following:
   - GREATER THAN OR EQUAL TO 6% hydrogen concentration in containment
   - Containment pressure above 60 psig
   - CONTAINMENT CLOSURE not established

Basis:

This EAL represents the inability to restore and maintain Reactor Vessel level above the top of active fuel. Fuel damage is probable if Reactor Vessel level cannot be restored, as available decay heat will cause boiling, further reducing the Reactor Vessel level. Setpoints enclosed in brackets (e.g., [30 ft], etc.) are used under adverse containment conditions [Ref. 3, 4].

This EAL is based on 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.* A number of variables– (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

In the cold shutdown mode, normal RCS level and Reactor Vessel level instrumentation systems will normally be available. In the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally available (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment High Range Monitor indication and possible alarm. Typical readings at full power are 1.0 to 2.5 R/hr. The Containment High Radiation indicators (1(2)RM126, 127 and 128) are log scaled with a span of $1\text{-}10^8$ R/hr. The 10 R/hr reading has been selected to be well above that expected under normal plant conditions [Ref. 1].

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (NIS N-31 and N-32) can be used as a tool for making such determinations.

. .

The GE is declared on the occurrence of the loss or imminent loss of function of <u>all three</u> barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause fuel clad failure. With the CONTAINMENT breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE.

CONTAINMENT CLOSURE is the action taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. CONTAINMENT CLOSURE should not be confused with refueling containment integrity as defined in technical specifications. Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS Barrier and Fuel Clad Barrier EALs, escalation to GE would not occur.

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 mixture of dissolved gasses in CONTAINMENT. However, CONTAINMENT monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

The containment design pressure (60 psig) is well in excess of that expected from the design basis loss of coolant accident [Ref. 7].

PBNP Basis Reference(s):

1. Eng Eval 2001-28, Containment High Radiation Channel Check Tolerance, 10/5/01
2. Volian Enterprises Calculation WEP-SPT-25
3. CSP-C.I UNIT 1 RED, CRITICAL SAFETY PROCEDURE SAFETY RELATED RESPONSE TO INADEQUATE CORE COOLING, Step 11
4. BG-CSP-Z.1, Response to High Containment Pressure
5. EPIP 10.3, POST-ACCIDENT CONTAINMENT HYDROGEN REDUCTION
6. FSAR Section 5.1, Containment System Structure
7. BG-CSP-ST.0, CSFST

Table E-0

Recognition Category E

Events Related to ISFSI Malfunction

INITIATING CONDITION MATRIX

	UE
EU1	Damage to a loaded cask CONFINEMENT BOUNDARY. *Op. Mode: Not Applicable*
EU2	Confirmed security event with potential loss of level of safety of the ISFSI *Op. Mode: Not Applicable*

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EU1

Initiating Condition -- UNUSUAL EVENT

Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: Not applicable

Emergency Action Level: (EU1.1 or EU1.2 or EU1.3)

EU1.1. Any one of the following natural phenomena events with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY.

Report of plant personnel of a:
- Tornado strike
- Earthquake
- Flood
- Lightning strike

EU1.2. Any of the following Accident conditions with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY.

- Vent Blockage
- Cask Drop
- Accidental Pressurization
- Air Vent and Outlet Shielding Reduction

EU1.3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask CONFINEMENT BOUNDARY.

Basis:

A UE in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

For EU1.1 and EU1.2, the results of the ISFSI Safety Analysis Report (SAR) and the related NRC Safety Evaluation Report were used to develop the site-specific list of natural phenomena events and accident conditions. These EALs address responses to events defined in the ISFSI Safety Analysis Report.

For EU1.3, any condition not explicitly detailed as an EAL threshold value, which, in the judgment of the Emergency Director, is a potential degradation in the level of safety of the ISFSI. Emergency Director judgment is to be based on known conditions and the expected response to mitigating activities within a short time period.

A UE in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. [Ref. 1, 2].

PBNP Basis Reference(s):

1. VSC-24, Conditions for Cask Use and Technical Specifications Docket No. 72-1007 Certificate of Compliance No. 1007
2. Standardized NUHOMS Horizontal Modular Storage System Certificate of Compliance No. 1004
3. Technical Specification 1.2.4 of the VSC-24 Certificate of Compliance
4. AOP-8G Ventilated Storage Cask (VSC) Drop or Tipover
5. Technical Specification 1.2.7 of the NUHOMS Certificate of Compliance
6. NUHOMS SAR
7. NUH-003 Rev 8, June 2004 "FSAR for Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel

EU2

Initiating Condition -- UNUSUAL EVENT

Confirmed Security Event with potential loss of level of safety of the ISFSI.

Operating Mode Applicability: Not applicable

Emergency Action Levels:

EU2.1. Security Event as determined from PBNP Physical Security Plan and reported by the Security Shift Supervisor.

Basis:

This EAL is based on PBNP Physical Security Plan Security events which do not represent a potential degradation in the level of safety of the ISFSI, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72.

Reference is made to the Security Shift Supervisor because these individuals are the designated personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Security Plan.

PBNP Basis Reference(s):

1. PBNP Physical Security Plan

Table F-0

Recognition Category F

Fission Product Barrier Degradation

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
FU1 ANY Loss or ANY Potential Loss of Containment *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FA1** ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FS1** Loss or Potential Loss of ANY Two Barriers *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FG1** Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*

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. UE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

   - At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

   - The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily lowering would represent an increasing risk to public health and safety.

2. Fission Product Barrier ICs must be capable of addressing event dynamics. Thus, the EAL Reference Table F-1 states that imminent (i.e., within 2 hours) Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

Deleted all BWR FPB Guidance

TABLE F-1

PBNP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
Conditions requiring entry into Core-Cooling RED Path (CSP-C.1)	Conditions requiring entry into Core Cooling-ORANGE Path (CSP-C.2) OR Conditions requiring entry into Heat Sink-RED Path (CSP-H.1)	Not Applicable	Conditions requiring entry into RCS Integrity-RED Path (CSP-P.1) OR Conditions requiring entry into Heat Sink-RED Path (CSP-H.1)	Not Applicable	Conditions requiring entry into Containment-RED Path (CSP-Z.1)
OR		**OR**		**OR**	
2. Primary Coolant Activity Level		**2. RCS Leak Rate**		**2. Containment Pressure**	
Coolant Activity GREATER THAN 300 µCi/gm I-131 equivalent	Not Applicable	GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling (LESS THAN OR EQUAL TO [80 degree F] 35 degree F)	Unisolable leak exceeding 60 gpm	Rapid unexplained lowering following initial rise OR Containment pressure or sump level response not consistent with LOCA conditions	60 PSIG and rising OR Hydrogen concentration in containment GREATER THAN OR EQUAL TO 6% OR Containment pressure GREATER THAN 25 psig with LESS THAN one full train of depressurization equipment operating
OR				**OR**	
3. Core Exit Thermocouple Readings				**3. Core Exit Thermocouple Reading**	

TABLE F-1

PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
GREATER THAN OR EQUAL TO 1200 degree F	GREATER THAN OR EQUAL TO 700 degree F			Not applicable	Core exit thermocouples in excess of 1200 degrees F and restoration procedures not effective within 15 minutes; OR, core exit thermocouples in excess of 700 degrees F with reactor vessel level below 27 ft RVLIS NR (with no RCPs running) top of active fuel and restoration procedures not effective within 15 minutes

TABLE F-1

PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
OR		**OR**		**OR**	
4. Reactor Vessel Water Level		**3. SG Tube Rupture**		**4. SG Secondary Side Release with P-to-S Leakage**	
Not Applicable	Level LESS THAN OR EQUAL TO ▪ 25 ft RVLIS NR (with no RCPs running) • [100 ft] 90 ft RVLIS WR (with 1 RCP running) ▪ [120 ft] 110 ft RVLIS WR (with 2 RCPs running)	SGTR that results in an ECCS (SI) Actuation	Not Applicable	RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate greater than 10 gpm with nonisolable steam release from affected S/G to the environment	Not applicable
				OR	
				5. CNMT Isolation Valves Status After CNMT Isolation	
				Containment isolation valve(s) not closed AND Downstream pathway to the environment exists, after containment isolation	Not Applicable
OR		**OR**		**OR**	
5. Containment Radiation Monitoring		**4. Containment Radiation Monitoring**		**6. Significant Radioactive Inventory in Containment**	

TABLE F-1

PBNP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
Containment rad monitor reading GREATER THAN 17 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128	Not Applicable	Containment rad monitor reading GREATER THAN 3.5 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128	Not Applicable	Not Applicable	Containment rad monitor reading GREATER THAN 15,900 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128

TABLE F-1

PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

*Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
OR		**OR**		**OR**	
6. Other Indications		**5. Other Indications**		**7. Other Indications**	
Failed Fuel Monitor (RE-109) reading GREATER THAN OR EQUAL TO 4500 mR/hr	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable
OR		**OR**		**OR**	
7. Emergency Director Judgment		**6. Emergency Director Judgment**		**8. Emergency Director Judgment**	
Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier		Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier		Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	

Basis Information For Table F-1
PBNP Emergency Action Level
Fission Product Barrier Reference Table

FUEL CLAD BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6 or 7)

The Fuel Clad Barrier is the zircalloy or stainless steel tubes that contain the fuel pellets.

1. Critical Safety Function Status

RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.

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. [Ref. 1, 2]

Heat Sink - RED indicates the ultimate heat sink function is under extreme challenge and thus these two items (Core Cooling – ORANGE or Heat Sink – RED) indicate potential loss of the Fuel Clad Barrier. CSP-H.1 is the Critical Safety Procedure that provides directions to respond to a loss of secondary heat sink in both steam generators. [Ref. 1, 3]

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

CSFST setpoints enclosed in brackets (e.g., [120 ft], etc.) are used under adverse containment conditions. Adverse containment conditions are defined as:

- Containment pressure is equal to or greater than 10 psig.
- Containment radiation is currently greater than or equal to 1E5 R/hr.
- Integrated dose is greater than 1E6 R or unknown.

The barrier loss/potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

2. Primary Coolant Activity Level

This value is 300 μCi/gm I_{131} equivalent. Assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no equivalent "Potential Loss" EAL for this item.

3. Core Exit Thermocouple Readings

Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked).

The "Loss" EAL 1200 degrees F reading should correspond to significant superheating of the coolant. This value corresponds to the temperature reading that indicates core cooling - RED in Fuel Clad Barrier EAL #1 which is 1200 degrees F. [Ref. 1]

The "Potential Loss" EAL 700 degrees F reading should correspond to loss of subcooling. This value corresponds to the temperature reading that indicates core cooling - ORANGE in Fuel Clad Barrier EAL #1 which is 700 degrees F. [Ref.1]

4. Reactor Vessel Water Level

There is no "Loss" EAL corresponding to this item because it is better covered by the other Fuel Clad Barrier "Loss" EALs.

The "Potential Loss" EAL is defined by the Core Cooling - ORANGE path [Ref.1]. The 25 ft water level is a consideration in the Core Cooling - ORANGE path only after the determination is made that no RCPs are running. With one RCP running, RVLIS WR value of [100 ft] 90 ft is used. With two RCPs running RVLIS WR values of [120 ft] 110 ft are used.

5. Containment Radiation Monitoring

The 17 R/hr reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere [Ref. 6,]. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier Loss EAL #4. Thus, this EAL indicates a loss of both the fuel clad barrier and a loss of RCS barrier.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

There is no "Potential Loss" EAL associated with this item.

6. Failed Fuel Monitor (RE-109) reading GREATER THAN 4500 mR/hr

A Failed Fuel Monitor reading of greater than 4500 mR/hr indicates the release of reactor coolant, with elevated activity indicative of fuel damage. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 activity into the Primary System. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage. [Ref. 8].

There is no "Potential Loss" EAL associated with this item

7. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. Such a determination should

include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

RCS BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6)

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

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

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. [Ref. 9, 10]

Heat Sink-Red path is entered if narrow range level in any S/G is equal to or less than [51%] 29% and total feedwater flow to S/Gs is equal to or less than 200 gpm. The combination of these two conditions indicates 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. [Ref. 11]

The barrier potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

There is no "Loss" EAL associated with this item.

2. RCS Leak Rate

The "Loss" EAL addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one charging pump discharging to the charging header. A second charging pump being required is indicative of a substantial RCS leak. 60 gpm is the minimum design flow rate for each charging pump. [Ref. 12]

3. SG Tube Rupture

This EAL is intended to address the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier "Loss" EAL #4 and Fuel Clad Barrier EALs. The "Loss" EAL addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation of ECCS (SI). ECCS (SI) actuation is caused by:

- PZR Low Pressure (equal to or less than 1735 psig)
- Steam Line Low Pressure (equal to or less than 530 psig)
- Containment High Pressure (equal to or greater than 5 psig)

140 gpm is the design maximum capacity of all charging pumps.

This is consistent to the RCS Barrier "Potential Loss" EAL #2. This condition is described by "entry into EOP-3 required by EOPs". By itself, this EAL will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment Barrier "Loss" EAL #4. [Ref. 14, 15]

There is no "Potential Loss" EAL.

4. Containment Radiation Monitoring

The 3.5 R/hr reading is a value which indicates the release of reactor coolant to the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the containment atmosphere [Ref. 6] This reading is less than that specified for Fuel Clad Barrier EAL #5. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL #5, fuel damage would also be indicated.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

There is no "Potential Loss" EAL associated with this item.

5. Other (Site-Specific) Indications

None

6. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

CONTAINMENT BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6 or 7 or 8)

The Containment Barrier includes the containment building, its 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.

1. Critical Safety Function Status

RED path indicates an extreme challenge to the safety function. Containment-Red path is entered if containment pressure is equal to or greater than 60 psig. This pressure is the containment design pressure and is well in excess of that expected from the design basis loss of coolant accident, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier. CSP-Z.1 is the Critical Safety Procedure that provides directions to respond to high containment pressure. [Ref. 16, 17, 18]

The barrier potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

There is no "Loss" EAL associated with this item.

2. Containment Pressure

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure rise indicates a loss of containment integrity. FSAR Figure 14.3.2-1 illustrates containment pressure response for a bounding LOCA. Containment pressure peaks at approximately 34 psia. [Ref. 25, 26, 27, 28]

Containment pressure or sump levels should rise as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure not rising indicates containment bypass and a loss of containment integrity.

The 60 PSIG for potential loss of containment is based on the containment design pressure. [Ref. 24, 25, 26, 27]

If hydrogen concentration reaches or exceeds 6% in an oxygen rich environment, an explosive mixture exists. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred As described above, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier. [Ref. 21, 22]

The second potential loss EAL represents a potential loss of containment in that the containment heat removal/depressurization system (but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint (25 psig) at which the equipment was supposed to have actuated. 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. [Ref. 16, 20, 23]

3. Core Exit Thermocouples

In this EAL, the restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is lowering or if the vessel water level is rising.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

The conditions in this potential loss EAL represent an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the Core Cooling and Heat Sink criteria in the Fuel and RCS barrier columns, this EAL would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path. [Ref. 1, 5]

There is no "Loss" EAL associated with this item.

4. SG Secondary Side Release With Primary To Secondary Leakage

This "loss" EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The first "loss" EAL addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers. In conjunction with RCS Barrier "loss" EAL #3, this would always result in the declaration of a Site Area Emergency. A faulted S/G means the existence of secondary side leakage that results in an uncontrolled lowering in steam generator pressure or the steam generator being completely depressurized. A ruptured S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection. Confirmation should be based on diagnostic activities consistent with EOP-0, Reactor Trip or Safety Injection.

The second "loss" EAL addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs. [Ref. 14]

It should be realized that the two "loss" EALs described above could be considered redundant. This was recognized during the development process. The inclusion of an EAL that uses

Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

A pressure boundary leakage of 10 gpm is used as the threshold in IC SU5.1, RCS Leakage, and is deemed appropriate for this EAL. For smaller breaks, not exceeding the normal charging capacity threshold in RCS Barrier "Potential Loss" EAL #2 (RCS Leak Rate) or not resulting in ECCS actuation in EAL #3 (SG Tube Rupture), this EAL results in a UE. For larger breaks, RCS barrier EALs #2 and #3 would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this EAL would exist in conjunction with RCS barrier "Loss" EAL #3 and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

There is no "Potential Loss" EAL associated with this item.

5. Containment Isolation Valve Status After Containment Isolation

This EAL is intended to address incomplete containment isolation that allows direct release to the environment. It represents a loss of the containment barrier.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no "Potential Loss" EAL associated with this item.

6. Significant Radioactive Inventory in Containment

The 15,900 R/hr reading is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS Barriers. [Ref. 6] A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

Monitors used for this fission product barrier potential loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

There is no "Loss" EAL associated with this item.

7. Other (Site-Specific) Indications

None

8. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

PBNP Basis Reference(s):

1. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
2. CSP-C.2, Response to Degraded Core Cooling
3. CSP-H.1, Response to Loss of Secondary Heat Sink
4. Volian Enterprises Calculation No. WEP-SPT-25, Reactor Vessel Level EOP Setpoints
5. CSP-C.1, Response to Inadequate Core Cooling
6. Calculation 2004-0006, Dose Rate Calculation for Containment High Range and Refueling Floor Area Radiation Monitors Under Accident Conditions
7. SAMG SAG-5, Reduce Fission Product Releases, Attachment D
8. Calc 2004-0008, Failed Fuel Monitor (RE-109) Reading / Fuel Damage Correlation
9. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 4
10. CSP-P.1, Response to Imminent Pressurized Thermal Shock Condition
11. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 3, Heat Sink

12. DBD-04, Chemical and Volume Control System, Section 3.9
13. BG-CSP-ST.0 Step ST-2, Critical Safety Status Trees
14. EOP-0, Reactor Trip Or Safety Injection
15. DBD-04, Chemical and Volume Control System
16. CSP-ST.0, Unit 1(2) Critical Safety Function Status Trees, Figure 5
17. BG-CSP-ST.0 Step ST-5
18. CSP-Z.1, Response to High Containment Pressure
19. FSAR Section 5.1, Containment System Structure
20. BG-CSP-ST.0, CSFST, Step F.0.5
21. CSP-C.1 Unit 1 Red, Critical Safety Procedure Safety Related Response To Inadequate Core Cooling, Step 11
22. EPIP 10.3, Post-Accident Containment Hydrogen Reduction
23. TS B 3.6.6, Containment Spray and Cooling Systems, pgs B 3.6.6-4 & -5, 10/20/02
24. FSAR Figure 14.3.2-1, Containment Pressure Curve used in PBNP BELOCA Analysis
25. FSAR Tables 14.3.2-1 through 14.3.2-3, Large Break Loss of Coolant Accident Analysis Data
26. FSAR 14.3.1, Small Break Loss of Coolant Accident Analysis
27. FSAR 14.3.2, Large Break Loss-Of-Coolant Accident Analysis
28. CSP-Z.1, Attachment B, Containment Isolation Valves
29. CALC WEP-SPT-12, EOP S1 Reduction Analysis 6/25/99
30. STPT 2.1, Safety Injection, Rev 2., dated 10/15/96

TABLE H-0

Recognition Category H

Hazards and Other Conditions Affecting Plant Safety

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
HU1 Natural and Destructive Phenomena Affecting the PROTECTED AREA. *Op. Modes: All*	**HA1** Natural and Destructive Phenomena Affecting the Plant VITAL AREA. *Op. Modes: All*		
HU2 FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection. *Op. Modes: All*	**HA2** FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown. *Op. Modes: All*		
HU3 Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant. *Op. Modes: All*	**HA3** Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Safety Systems Required to Establish or Maintain Safe Shutdown. *Op. Modes: All*		
HU4 Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant. *Op. Modes: All*	**HA4** Confirmed Security Event in a Plant PROTECTED AREA. *Op. Modes: All*	**HS1** Confirmed Security Event in a Plant VITAL AREA. *Op. Modes: All*	**HG1** Security Event Resulting in Loss Of Physical Control of the Facility. *Op. Modes: All*
HU5 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE. *Op. Modes: All*	**HA6** Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. *Op. Modes: All*	**HS3** Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency. *Op. Modes: All*	**HG2** Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency. *Op. Modes: All*
	HA5 Control Room Evacuation Has Been Initiated. *Op. Modes: All*	**HS2** Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established. *Op. Modes: All*	

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HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Initiating Condition – UNUSUAL EVENT

Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Level: (HU1.1 or HU1.2 or HU1.3 or HU1.4 or HU1.5 or HU1.6or HU1.7)

HU1.1. Earthquake felt in plant as indicated by:

Activation of 2 or more seismic monitors

AND

Verified by:

- Actual ground shaking

  OR

- By contacting the U.S. Geological Survey National Earthquake Information Center

HU1.2. Report by plant personnel of tornado or high winds GREATER THAN 108 mph (15 minute average) striking within PROTECTED AREA boundary.

HU1.3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary.

HU1.4. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

HU1.5. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

HU1.6. Uncontrolled flooding in the following areas of the plant that has the potential to affect safety related equipment needed for the current operating mode:

- Auxiliary building caused by rupture of the SW header

  OR

- Water intake structure caused by rupture of a circulating water system expansion joint or fire water main.

HU1.7. Lake (forebay) level GREATER THAN OR EQUAL TO 8.0 ft (587.2 ft IGLD)

Basis:

UE in this IC are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the EALs define the location of the event based on the potential for damage to equipment contained therein. Escalation of the event to an Alert occurs when the magnitude of the event is sufficient to result in damage to equipment contained in the specified location.

HU1.1. Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. Method of detection is based on instrumentation, validated by a reliable source (U.S. Geological Survey National Earthquake Information Center), or operator assessment [Ref. 1, 2, 3, 4]. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a *"felt earthquake"* is:

> An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01g.

HU1.2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind site specific value is based on the FSAR design basis sustained wind speed of 108 mph. Wind loads of this magnitude can cause damage to safety functions. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert. [Ref. 5, 6]

HU1.3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. If the crash is confirmed to affect a plant VITAL AREA, the event may be escalated to Alert. [Ref. 6]

For HU1.4 only those EXPLOSIONs of sufficient force to damage permanent structures or equipment within the PROTECTED AREA should be considered. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The Emergency director also needs to consider any security aspects of the EXPLOSION, if applicable. [Ref. 6]

HU1.5 is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIREs and flammable gas build up are appropriately classified via HU2 and HU3. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant. This EAL is consistent with the definition of a UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs. [Ref. 8, 9]

HU1.6 addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The auxiliary building and water intake structure are the vulnerable areas indicated in the IPE that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged. Escalation of the

emergency classification is based on the damage caused or by access restrictions that prevent necessary plant operations or systems monitoring. [Ref. 11, 12]

HU1.7 covers other site-specific phenomena such as flood, or seiche. This EAL can also be precursors of more serious events. Lake water level GREATER THAN OR EQUAL TO 8 feet (587.2 ft elevation IGLD) corresponds to the Turbine Building floor elevation. Both the Turbine Building and Circ Water Pumphouse are specifically susceptible to external flooding.

PBNP Basis Reference(s):

1. AOP-28 Seismic Event
2. FSAR Section 2.9 Seismology
3. STPT 22.1 Seismic Event Monitoring
4. EPRI document, "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989
5. FSAR Section 2.6, Meteorology
6. Bechtel Drawing C-3 Plant Areas
7. ARP 1(2) C33 1-2 Hydrogen Pressure High-Low Alarm
8. ARP 1(2) C33 1-3 Hydrogen Supply Pressure Low Alarm
9. AOP-5A Loss of Condenser Vacuum
10. NPC95-00559, PBNP Individual Plant Examination (IPE) for internal events and internal flood.
11. DG-C02, Internal Flooding
12. DBD-T41, Module A, "Hazards-Internal and External Flooding"
13. FSAR Section 2.5, Hydrology
14. International Great Lakes Datum (IGLD)

HU2

Initiating Condition -- UNUSUAL EVENT

FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.

Operating Mode Applicability: All

Emergency Action Level:

HU2.1. FIRE in buildings or areas contiguous to any of the Table H-1 areas not extinguished within 15 minutes of control room notification or verification of a control room alarm

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

Basis:

The purpose of this IC is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, Detection is visual observation and report by plant personnel or sensor alarm indication. The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. Verification of a fire detection system alarm includes actions that can be taken with the control room or other nearby site-specific location to ensure that the alarm is not spurious. A verified alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket). This EAL excludes waste-basket FIREs and other small FIREs of no safety consequence.

Escalation to a higher emergency class is by IC HA4, "FIRE Affecting the Operability of Plant Safety Systems Required for the Current Operating Mode".

PBNP Basis Reference(s):

1. Bechtel Drawing C-3 Plant Areas

HU3

Initiating Condition –UNUSUAL EVENT

Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.

Operating Mode Applicability: All

Emergency Action Levels: (HU3.1 or HU3.2)

HU3.1. Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.

HU3.2. Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

This IC is based on the existence of uncontrolled releases of toxic or flammable gas that may enter the site boundary and affect normal plant operations. It is intended that releases of toxic or flammable gases are of sufficient quantity, and the release point of such gases is such that normal plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. The EALs are intended to not require significant assessment or quantification. The IC assumes an uncontrolled process that has the potential to affect plant operations, or personnel safety.

Escalation of this EAL is via HA3, which involves a quantified release of toxic or flammable gas affecting VITAL AREAs.

PBNP Basis Reference(s):

None

HU4

Initiating Condition –UNUSUAL EVENT

Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant.

Operating Mode Applicability: All

Emergency Action Levels: (HU4.1 or HU4.2)

HU4.1. Security Shift Supervisor reports ANY of the following:
- Suspected sabotage device discovered within the plant Protected Area
- Suspected sabotage device discovered outside the Protected Area or in the plant switchyard
- Confirmed tampering with safety-related equipment
- A hostage situation that disrupts NORMAL PLANT OPERATIONS
- Civil disturbance or strike which disrupts NORMAL PLANT OPERATIONS
- Internal disturbance that is not a short lived or that is not a harmless outburst involving ANY individuals within the Protected Area
- Malevolent use of a vehicle outside the Protected Area which disrupts NORMAL PLANT OPERATIONS

HU4.2 Notification of a credible site-specific threat by the Security Shift Supervisor or outside agency (e.g., NRC, military or law enforcement)

Basis:

Reference is made to the Security Shift Supervisor because this individual is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan.

HU4.1 is based on Physical Security Plan. Security events which do not represent a potential degradation in the level of safety of the plant, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Examples of security events that indicate Potential Degradation in the Level of Safety of the Plant are provided for consideration.

INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE would result in EAL escalation to an ALERT.

The intent of HU4.2 is to ensure that appropriate notifications for the security threat are made in a timely manner. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the Physical Security Plan.

A higher initial classification could be made based upon the nature and timing of the threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification in accordance with the Physical Security Plan and Emergency Plans.

PBNP Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. PBNP Physical Security Plan
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. NMC fleet Security Threat Assessment Policy

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition –UNUSUAL EVENT

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE.

Operating Mode Applicability: All

Emergency Action Level:

HU5.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Basis:

This EAL is intended to address 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 UE emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency response procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

PBNP Basis Reference(s):

None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition – ALERT

Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability: All

Emergency Action Levels: (HA1.1 or HA1.2 or HA1.3 or HA1.4 or HA1.5 or HA1.6)

HA1.1. Seismic event GREATER THAN Operating Basis Earthquake (OBE) as indicated by:

VALID seismic monitor indication of ground acceleration **EITHER:**
GREATER THAN OR EQUAL TO 0.06 g horizontal

OR

GREATER THAN OR EQUAL TO 0.04 g vertical

HA1.2. Tornado or high winds (15 minute average) GREATER THAN 108 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures / equipment or Control Room indication of degraded performance of those systems.

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

HA1.3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or control room indication of degraded performance of those systems:

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

HA1.4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas:

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

HA1.5. Uncontrolled flooding in following areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

Auxiliary building caused by rupture of the SW header

OR

Water intake structure caused by rupture of a circulating water system expansion joint or fire water main.

HA1.6 Lake (forebay) level GREATER THAN OR EQUAL TO 9.0 ft (588.2 ft IGLD)

Basis:

The EALs in this IC escalate from the UE EALs in HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other ICs (e.g., System Malfunction).

HA1.1 is based on the FSAR operating basis earthquake (OBE) of 0.06 g horizontal or 0.04 g vertical acceleration. Seismic events of this magnitude can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. [Ref. 1, 2, 3, 4]

HA1.2 is based on the FSAR design basis sustained wind speed of 108 mph. Wind loads of this magnitude can cause damage to safety functions.

Wind speed is measured as the 15 minute average wind speed. This EAL addresses events that may have resulted in a plant VITAL AREA being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant structures, systems or equipment. [Ref. 5, 6] The list of plant structures/equipment contains functions required for safe shutdown of the plant. Ultimate Heat Sink is not included in this list as there are no events identified which would adversely impact Lake Level and the system pumps and associated equipment which are located in the Intake Building and other listed structures.

HA1.3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. The

list of plant structures/equipment contains functions required for safe shutdown of the plant. Ultimate Heat Sink is not included in this list as there are no events identified which would adversely impact Lake Level and the system pumps and associated equipment, which are located in the Intake Building and other listed structures.

HA1.4 is intended to address the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant. [Ref. 7, 8, 9] The list of plant structures/equipment containing functions required for safe shutdown of the plant. Ultimate Heat Sink is not included in this list as there are no events identified which would adversely impact Lake Level and the system pumps associated equipment, which are located in the Intake Building and other listed structures.

HA1.5 addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The auxiliary building and water intake structure are those areas identified in the IPE that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged . [Ref. 10]

HA1.6 covers high lake (forebay) 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 GREATER THAN OR EQUAL TO 9.0 feet (588.2 ft elevation IGLD) corresponds to that which can result in flooding of vital areas containing safe shutdown equipment.

PBNP Basis Reference(s):

1. AOP-28 Seismic Event
2. FSAR Section 2.9 Seismology
3. STPT 22.1 Seismic Event Monitoring
4. EPRI document, "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989
5. FSAR Section 2.6, Meteorology
6. Bechtel Drawing C-3 Plant Areas
7. ARP 1(2) C33 1-2 Hydrogen Pressure High-Low Alarm
8. ARP 1(2) C33 1-3 Hydrogen Supply Pressure Low Alarm
9. AOP-5A Loss of Condenser Vacuum
10. NPC95-005559 PBNP Individual Plant Examination (IPE) for internal events and internal flood.
11. FSAR Section 2.5, Hydrology
12. International Great Lakes Datum (IGLD)

HA2

Initiating Condition -- ALERT

FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

Emergency Action Level:

HA2.1. FIRE or EXPLOSION in any of the following areas (Table H-1):

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

AND

Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.

Basis:

These areas contain systems and components required for the safe shutdown functions of the plant. The PBNP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode. This will make it easier to determine if the FIRE or EXPLOSION is potentially affecting one or more redundant trains of safety systems. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs.

This EAL addresses a FIRE / EXPLOSION and not the degradation in performance of affected systems. System degradation is addressed in the System Malfunction EALs. The reference to damage of systems is used to identify the magnitude of the FIRE / EXPLOSION and to discriminate against minor FIREs / EXPLOSIONs. The reference to safety systems is included to discriminate against FIREs / EXPLOSIONs in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE / EXPLOSION was large enough to cause damage to these systems.
This situation is not the same as removing equipment for maintenance that is covered by a plant's Technical Specifications. Removal of equipment for maintenance is a planned activity controlled in accordance with procedures and, as such, does not constitute a substantial degradation in the level of safety of the plant. A FIRE / EXPLOSION is an UNPLANNED activity and, as such, does

constitute a substantial degradation in the level of safety of the plant. In this situation, an Alert classification is warranted.

The inclusion of a "report of VISIBLE DAMAGE" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform these damage assessments. The Emergency Director also needs to consider any security aspects of the EXPLOSIONs, if applicable.

PBNP Basis Reference(s):

None

HA3

Initiating Condition -- ALERT

Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

Emergency Action Levels: (HA3.1 or HA3.2)

HA3.1. Report or detection of toxic gases within or contiguous to a VITAL AREA (Table H-1) in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

HA3.2. Report or detection of gases in concentration GREATER THAN the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA (Table H-1).

Table H-1 VITAL AREA
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

Basis:

This IC is based on gases that affect the safe operation of the plant. This IC applies to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas. The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to plant VITAL AREAs. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radioactive Effluent, or Emergency Director Judgment ICs.

HA3.1 is met if measurement of toxic gas concentration results in an atmosphere that is IDLH within a VITAL AREA or any area or building contiguous to VITAL AREA. Exposure to an IDLH atmosphere will result in immediate harm to unprotected personnel, and would preclude access to any such affected areas.

HA3.2 is met when the flammable gas concentration in a VITAL AREA or any building or area contiguous to a VITAL AREA exceed the LOWER FLAMMABILITY LIMIT. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL addresses concentrations at which gases can ignite/support combustion. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Once it has been determined that an uncontrolled release is occurring, then sampling must be done to determine if the concentration of the released gas is within this range.

PBNP Basis Reference(s):

None

HA4

Initiating Condition -- ALERT

Confirmed Security Event in a Plant PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Levels: (HA4.1 or HA4.2)

HA4.1. INTRUSION into the PROTECTED AREA by a HOSTILE FORCE.

HA4.2. Security Shift Supervisor reports any of the following:
- Sabotage device discovered in the PROTECTED AREA
- Standoff attack on the site protected area by a HOSTILE FORCE (i.e., Sniper)
- ANY Security event of increasing severity that persists for GREATER THAN 30 minutes:
  - Credible bomb threats
  - Extortion
  - Suspicious Fire or Explosion
  - Significant Security System Hardware Failure
  - Loss of Guard Post Contact

Basis:

This class of security events represents an escalated threat to plant safety above that contained in the UE. A confirmed INTRUSION report is satisfied if physical evidence indicates the presence of a HOSTILE FORCE within the PROTECTED AREA.

The Physical Security Plan identifies numerous events/conditions that constitute a threat/compromise to station security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered.

INTRUSION into a VITAL AREA by a HOSTILE FORCE will escalate this event to a Site Area Emergency.

Reference is made to Security Shift Supervisor because this individual is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy

HA5

Initiating Condition -- ALERT

Control Room Evacuation Has Been Initiated.

Operating Mode Applicability: All

Emergency Action Level:

HA5.1. Entry into AOP-10 Control Room Inaccessibility for control room evacuation.

Basis:

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facility is necessary. The AOP-10 series of procedures provide specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations. Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

PBNP Basis Reference(s):

1. AOP-10 Control Room Inaccessibility
2. AOP-10A Safe Shutdown - Local Control
3. AOP-10B U1(2) Safe to Cold Shutdown in Local Control

HA6

Initiating Condition – ALERT

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.

Operating Mode Applicability: All

Emergency Action Level:

HA6.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

This EAL is intended to address 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 Alert emergency class.

PBNP Basis Reference(s):

None

HS1

Initiating Condition – SITE AREA EMERGENCY

Confirmed Security Event in a Plant VITAL AREA.

Operating Mode Applicability: All

Emergency Action Levels: (HS1.1 or HS1.2)

HS1.1. INTRUSION into the plant VITAL AREA by a HOSTILE FORCE.

HS1.2. Security Supervision reports confirmed sabotage discovered in a VITAL AREA

Basis:

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC in that a HOSTILE FORCE has progressed from the PROTECTED AREA to a VITAL AREA.

Consideration is given to the following types of events when evaluating an event against the criteria of the site specific Physical Security Plan: SABOTAGE and HOSTAGE / EXTORTION. The Physical Security Plan identifies numerous events/conditions that constitute a threat/compromise to a Station's security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered.

Loss of Plant Control would escalate this event to a GENERAL EMERGENCY.

Reference is made to Security Shift Supervisor because this individual is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan.

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy

HS2

Initiating Condition – SITE AREA EMERGENCY

Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.

Operating Mode Applicability: All

Emergency Action Level:

HS2.1. Control room evacuation has been initiated.

AND

Control of the plant cannot be established per AOP-10A Safe Shutdown - Local Control within 15 minutes.

Basis:

Expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the time for transfer that the operator has control of the plant from the remote shutdown panel.

The AOP-10 series of procedures provide specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

Escalation of this event, if appropriate, would be by Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Emergency Director Judgment ICs.

PBNP Basis Reference(s):

1. AOP-10 Control Room Inaccessibility
2. AOP-10A Safe Shutdown - Local Control
3. AOP-10B Safe to Cold Shutdown in Local Control
4. BG AOP-10A, Safe Shutdown – Local Control

HS3

Initiating Condition – SITE AREA EMERGENCY

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.

Operating Mode Applicability: All

Emergency Action Level:

HS3.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

This EAL is intended to address 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 class description for Site Area Emergency.

PBNP Basis Reference(s):

None

HG1

Initiating Condition – GENERAL EMERGENCY

Security Event Resulting in Loss Of Physical Control of the Facility.

Operating Mode Applicability: All

Emergency Action Level:

HG1.1. A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of **EITHER**:

A VITAL AREA such that operation of equipment required for safe shutdown is lost

OR

Spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in the pool).

Basis:

This IC encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAs (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location. Typically, these safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink). If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the above initiating condition is not met.

This EAL addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in pool).

Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se.

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy

HG2

Initiating Condition – GENERAL EMERGENCY

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.

Operating Mode Applicability: All

Emergency Action Level:

HG2.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL is intended to address 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 General Emergency class.

PBNP Basis Reference(s):

None

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Table S-0

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU1 Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA5** AC Power Capability to Essential Busses Reduced to a Single Power Source for GREATER THAN 15 minutes Such That Any Additional Single Failure Would Result in Station Blackout. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS1** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SG1** Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*
	SA2 Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was Successful. *Op. Modes: Power Operation, Startup, Hot Standby*	**SS2** Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was NOT Successful. *Op. Modes: Power Operation, Startup*	**SG2** Failure of the Reactor Protection System to Complete an Automatic Trip and Manual Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core. *Op. Modes: Power Operation, Startup*
SU2 Inability to Reach Required Shutdown Within Technical Specification Limits. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA3** *Deleted*	**SS4** Complete Loss of Heat Removal Capability. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	
SU3 UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN 15 Minutes *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA4** UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS6** Inability to Monitor a SIGNIFICANT TRANSIENT in Progress. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

SU7 *Deleted*

SU4 Fuel Clad Degradation.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU5 RCS Leakage.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU8 Inadvertent Criticality.
Op Modes: Hot Standby, Hot Shutdown

SA1 *Deleted*

SS3 Loss of All Vital DC Power.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SS5 *Deleted*

SYSTEM MALFUNCTION

SU1

Initiating Condition -- UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SU1.1. Loss of all **offsite** power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN 15 minutes.

AND

At least 1 emergency generator is supplying power to an emergency bus.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The 4160 VAC system includes two safety-related (essential) buses per unit, 1(2)-A05 (A train) and 1(2)-A06 (B train). Offsite power from the 345 KVAC system is stepped down through the 13.8 KVAC system to the Low Voltage Station Auxiliary Transformer (LVSAT) 1(2)-X04. The LVSATs provide power to 4160 VAC switching buses 1(2)-A03 and 1(2)-A04, which in turn provide power to safety-related buses 1(2)-A05 and 1(2)-A06.

During emergency or abnormal situations, the 4160 VAC system is supplied by emergency diesel generators (G01 through G04) or the gas turbine generator (G05). Following a loss of power, ECA 0.0 provides guidance to restore power to any 4160 VAC safety-related bus. For the purpose of classification under this EAL, offsite power sources include any of the following:

- 345 KVAC system supplying power to the 13.8 KVAC system and the unit LVSATs
- Cross-tying with the opposite unit power supply
- Backfeeding power to the UATs through the 19 KVAC system and the main step-up transformer X-01. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source.

PBNP has the capability to cross-tie AC power from the other unit and therefore takes credit for the redundant power source for this IC. However, the inability to effect the cross-tie within 15 minutes warrants declaring a UE.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A Main Power Transformer Backfeed

SU2

Initiating Condition -- UNUSUAL EVENT

Inability to Reach Required Shutdown Within Technical Specification Limits.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SU2.1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the PBNP Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate UE is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of a UE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. PBNP Technical Specifications

SYSTEM MALFUNCTION

SU3

Initiating Condition – UNUSUAL EVENT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN 15 Minutes

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SU3.1. UNPLANNED loss of most or all annunciators or indicators associated with the following safety systems for GREATER THAN 15 minutes

- ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This is addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

The specified annunciators or indicators for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no IC is indicated during these modes of operation.

This UE will be escalated to an Alert if a transient is in progress during the loss of annunciation or indication.

PBNP Basis Reference(s):

1. OM 1.1, Conduct of Plant Operations, Attachment 2

Initiating Condition -- UNUSUAL EVENT

Fuel Clad Degradation.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Levels: (SU4.1 or SU4.2)

SU4.1. Failed Fuel Monitor (RE-109) GREATER THAN 750 mR/hr indicating fuel clad degradationgreater than Technical Specification allowable limits

SU4.2. Coolant sample activity GREATER THAN 50 µCi/gm dose equivalent I-131 indicating fuel clad degradation greater than Technical Specification allowable limits.

Basis:

This IC is included as a UE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. SU4.1 addresses the failed fuel monitor, that provides indication of fuel clad integrity [Ref. 2]. 750 mR/hr is the value that corresponds to approximately 0.1% fuel clad damage. SU4.2 addresses coolant samples exceeding coolant technical specifications for iodine spike [Ref. 1]. Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs. The referenced Technical Specification limits are mode dependent.

PBNP Basis Reference(s):

1. Tech Spec 3.4.16 – RCS Specific Activity
2. Calc 2004-0019, Failed Fuel Monitor (RE-109) Reading / Fuel Damage Correlation

Initiating Condition -- UNUSUAL EVENT

RCS Leakage.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Levels: (SU5.1 or SU5.2)

SU5.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm.

SU5.2. Identified leakage GREATER THAN 25 gpm.

Basis:

This IC is included as a UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. TS 3.4.13, RCS Operational Leakage limits
2. OI 55, Primary Leak Rate Calculation
3. OM 3.19, Reactor Coolant System Leakage Determination

SYSTEM MALFUNCTION

SU6

Initiating Condition -- UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Levels: (SU6.1 or SU6.2)

SU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Plant Public Address System
• Security Radio
• Commercial Phone System
• Portable radios
• Sound power phones

SU6.2. Loss of all Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• Emergency Notification System (ENS)
• Health Physics Network (HPN)
• Operations Control Counterpart Link (OCCL)
• Management Counterpart Link (MCL)
• Protective Measures Counterpart Link (PMCL)
• Reactor Safety Counterpart Link (RSCL)
• Nuclear Accident Reporting System (NARS)
• Commercial Phone System
• General Telephone Lines
• Manitowoc County Sheriff's Department Radio

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary

means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

Table C-1 onsite communications loss [Ref. 1, 2] encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system (Gaitronics) and radios / walkie talkies).

Table C-2 offsite communications loss [Ref. 1, 2] encompasses the loss of all means of communications with offsite authorities. This includes the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

PBNP Basis Reference(s):

1. EPMP 2.1, Testing of Communications Equipment
2. EPMP 2.1A, Monthly Communications Test

SU8

Initiating Condition – UNUSUAL EVENT

Inadvertent Criticality.

OPERATING MODE APPLICABILITY Hot Standby
Hot Shutdown

Emergency Action Level: (SU8.1)

SU8.1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses inadvertent criticality events. While the primary concern of this IC is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling IC is CU8.

This condition can be identified using the startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements such as shutdown bank withdrawal. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

This condition can be identified using startup rate monitors (1(2)NI-31D/32D - Source Range Startup Rate, and 1(2)NI-35D/36D - Intermediate Range Startup Rate) [Ref. 1].

Escalation would be by the Fission Product Barrier Matrix, as appropriate to the operating mode at the time of the event, or by Emergency Director Judgment.

Note: This EAL is SU8 following SU6. SU7 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

PBNP Basis Reference(s):

1. OP 1B, Reactor Startup

Initiating Condition – ALERT

Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was Successful.

Operating Mode Applicability: Power Operation
Startup
Hot Standby

Emergency Action Level:

SA2.1. Indication(s) exist that a Reactor Protection System (RPS) setpoint was exceeded

AND

RPS automatic trip did NOT reduce power to LESS THAN 5%

AND

Any of the following operator actions are successful in reducing power to LESS THAN 5%:

- Use of Reactor Trip Buttons
- De-energizing 1(2)B01 and 1(2)B02

Basis:

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS integrity. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue. A manual trip is any set of actions by the reactor operator(s) in the control room which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip button). Failure of manual trip would escalate the event to a Site Area Emergency.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees
2. BG-CSP-ST.0, Critical Safety Function Status Trees
3. EOP-0, Reactor Trip or Safety Injection

Initiating Condition -- ALERT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SA4.1. UNPLANNED loss of most or all annunciators or indicators associated with the following safety systems for GREATER THAN 15 minutes

- ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

AND

Either of the following: (a or b)

a. A SIGNIFICANT TRANSIENT is in progress.

OR

b. Compensatory non-alarming indications are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

SIGNIFICANT TRANSIENT includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Compensatory non-alarming indications include the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

The specified annunciators or indicators for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no IC is indicated during these modes of operation.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress.

Note: This EAL is SA4 following SA2. SA2 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

PBNP Basis Reference(s):

1. OM 1.1, Conduct of Plant Operations, Attachment 2

Initiating Condition – ALERT

AC power capability to essential busses reduced to a single power source for GREATER THAN 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SA5.1. AC power capability to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 reduced to only one of the following sources for GREATER THAN 15 minutes

- A single emergency diesel generator (G01, G02, G03 or G04)
- LVSAT 1(2)-X04
- Cross-tying with the opposite unit power supply

AND

Any additional single failure will result in station blackout.

Basis:

This IC and the associated EALs are intended to provide an escalation from IC SU1, "Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes." The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency diesel generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency diesels with only one train of emergency busses being backfed from the unit main generator, or the loss of onsite emergency diesels with only one train of emergency busses being backfed from offsite power. Offsite power sources include the four 345 KVAC lines (111, 121, Q303 and 151) through the 13.8 KVAC system to the LVSAT and 345 KVAC backfed through the 19 KVAC system to the UAT. Note that the time required to effect a backfeed to the UAT is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source. Onsite power sources consist of the emergency diesel generators, the gas turbine generator (G05) feeding the 13.8 KVAC system to the LVSATs, the unit main turbine generator, and power supplied from the opposite unit. Several combinations of power failures could therefore satisfy this EAL. The subsequent loss of the single remaining power source would escalate the event to a Site Area Emergency in accordance with IC SS1, "Loss of All Offsite and Loss of All Onsite AC Power to Essential Busses."

PBNP Basis Reference(s):

PBNP

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A Unit 1, Main Power Transformer Backfeed

SS1

Initiating Condition -- SITE AREA EMERGENCY

Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SS1.1. Loss of all offsite power to safety-related 4160 VAC buses 1(2)-A05 AND 1(2)-A06

AND

Failure of all emergency generators to supply power to safety-related 4160 VAC buses.

AND

Failure to restore power to at least one safety-related 4160 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency.

Offsite power sources include the four 345 KVAC lines (111, 121, Q303 and 151) through the 13.8 KVAC system to the LVSAT and 345 KVAC backfed through the 19 KVAC system to the UAT. Note that the time required to effect a backfeed to the UAT is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source. Onsite power sources consist of the emergency diesel generators, the gas turbine generator (G05) feeding the 13.8 KVAC system to the LVSATs, the unit main turbine generator, and power supplied from the opposite unit.

Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power."

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 this bus was the only energized bus then a Site Area Emergency per SS1 should be declared.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A U1, Main Power Transformer Backfeed

Initiating Condition – SITE AREA EMERGENCY

Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was NOT Successful.

Operating Mode Applicability: Power Operation
Startup

Emergency Action Level:

SS2.1. Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%.

Basis:

Automatic and manual trip are not considered successful if action away from the reactor control console was required to trip the reactor.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Emergency Director Judgment ICs.

Automatic or manual reactor trip is considered successful if actions taken at the main control panels (use of reactor trip buttons, de-energizing 1(2)B01 and 1(2)B02) result in reducing reactor power less than 5%.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees
2. BG-CSP-ST.0, CRITICAL SAF ETY FUNCTION STATUS TREES
3. CSP-S.1, Response to Nuclear Power Generation/ATWS4. EOP-0, Reactor Trip or Safety Injection

Initiating Condition – SITE AREA EMERGENCY

Loss of All Vital DC Power.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SS3.1. Loss of all vital DC power based on LESS THAN 115 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for GREATER THAN 15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation, or Emergency Director Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The safety-related station batteries have been sized to carry their expected shutdown loads following a plant trip/LOCA and loss of offsite power or following a station blackout for a period of one hour without battery terminal voltage falling below the minimum bus voltages required for equipment operation. The bus voltage for full operability at all connected loads varies and is specific for each bus. At 115 VDC loss of equipment function would begin to occur. 105 VDC was not selected because it is the voltage of a completely discharged battery. By the time the bus voltage has degraded to below 105 VDC, the ability to monitor and control important plant safety functions would be severely compromised. 115 VDC represents a value at which some plant functions would begin to be lost. 115 VDC is below the minimum normal operating voltage, to preclude entering this EAL due to normal operational situations.

PBNP Basis Reference(s):

1. 0-SOP-DC-001/2/3/4

SS4

Initiating Condition -- SITE AREA EMERGENCY

Complete Loss of Heat Removal Capability.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SS4.1. Loss of core cooling (CSP-C.1) AND heat sink (CSP-H.1).

Basis:

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to General Emergency would be via Abnormal Rad Levels / Radiological Effluent, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. CSP-C.1, Response to Inadequate Core Cooling
2. CSP-H.1, Response to Loss of Secondary Heat Sink

Initiating Condition -- SITE AREA EMERGENCY

Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SS6.1. Loss of most or all annunciators associated with the following safety systems

- ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

AND

SIGNIFICANT TRANSIENT in progress.

AND

Compensatory non-alarming indications are unavailable.

AND

Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, or maintain containment intact are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the inability of the control room staff to monitor the plant response to a transient. A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public.

SIGNIFICANT TRANSIENT includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Compensatory non-alarming indications include the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room.

Indications needed to monitor safety functions necessary for protection of the public include control room indications, computer generated indications and dedicated annunciation capability. The specific indications are those used to monitor the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

"Planned" and "UNPLANNED" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification of the number of annunciators and indicators is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.**PBNP Basis Reference(s):**

OM 1.1, Conduct of Plant Operations, Attachment 2

Initiating Condition -- GENERAL EMERGENCY

Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SG1.1. Loss of all **offsite** power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06

AND

Failure of all emergency diesel generators to supply power to safety-related 4160 VAC buses.

AND

Either of the following: (a or b)

a. Restoration of at least one safety-related 4160 VAC bus within 4 hours is NOT likely

OR

b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by conditions requiring entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. The 4 hours to restore AC power is based on the site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout," [Ref 10]. Appropriate allowance for offsite emergency response including evacuation of surrounding areas should be considered. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.
In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent? (Refer to Table F-1 for more information.)

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

PBNP Basis Reference(s):

1. DBD-T-46, Section 3.1, Station Blackout
2. 10 CFR 50.63 and Regulatory Guide 1.155, Station Blackout
3. DBD-22, 4160 VAC System
4. DBD-18, 13.8 KVAC System
5. ECA 0.0, Loss of All AC Power
6. FSAR Section 8, Electrical Systems
7. AOP-14A U1, Main Power Transformer Backfeed
8. CSP-C.1, Response to Inadequate Core Cooling
9. CSP-C.2, Response to Degraded Core Cooling
10. FSAR Appendix A, "Station Blackout"

SG2

Initiating Condition -- GENERAL EMERGENCY

Failure of the Reactor Protection System to Complete an Automatic Trip and Manual Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability: Power Operation
Startup

Emergency Action Level:

SG2.1. Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%.

AND

Either of the following: (a or b)

a. Indication(s) exists that the core cooling is extremely challenged Core Cooling - RED path (CSP-C.1).

OR

b. Indication(s) exists that heat removal is extremely challenged Heat Sink - RED path (CSP-H.1).

Basis:

Automatic and manual trip are not considered successful if action away from the reactor control console is required to trip the reactor.

Under the conditions of this IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console, such as emergency boration, the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence.

The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F or that the reactor vessel water level is below the top of active fuel. This EAL equates to a core cooling RED condition and an entry into a critical safety procedure (CSP-C.1).

Another consideration is the inability to initially remove heat during the early stages of this sequence. If emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist.

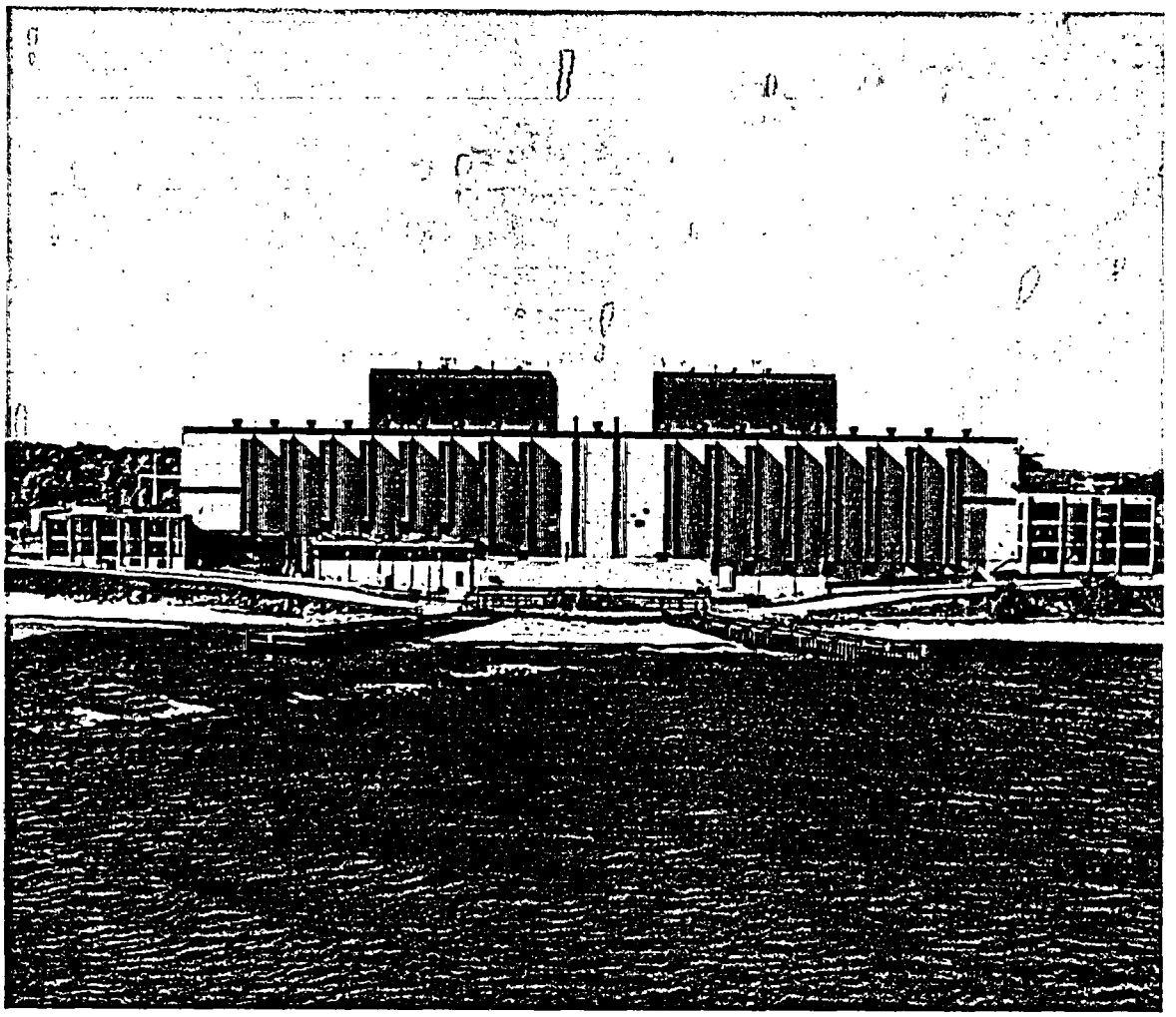
This EAL equates to a Heat Sink RED condition and an entry into a critical safety procedure (CSP-H.1).

In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design (typically 3 to 5% power) a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees, Figures 1, 2 and 3
2. CSP-S.1, Response to Nuclear Power Generation/ATWS
3. CSP-C.1, Response to Inadequate Core Cooling
4. CSP-H.1, Response to Loss of Secondary Heat Sink

ENCLOSURE IV
ATTACHMENT 3



POINT BEACH
REVISED EAL SUBMITTAL

ATTACHMENT 3: EMERGENCY PLAN AND PROCEDURE CHANGES

Enclosure IV

Attachment 3: Emergency Plan and Procedure Changes

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL

Note: Refer to NP 1.1.3 for requirements.

I - INITIATION

Doc Number EP Appendix B Unit PB0 Usage Level Reference Proposed Rev No 22

Title Emergency Classification Classification NNSR

☒ Revision ☐ Cancellation ☐ New Document ☐ Other (e.g., periodic review, admin hold) ____

List Temporary Changes/Feedbacks Incorporated: CAP 050434, CAP050594

Description of Alteration/Reason (If necessary, continue description of changes on PBF-0026c and attach.)

Change to the NEI 99-01 scheme - Total Rewrite - from the NUREG

List other documents required to be effective concurrently with the revision (e.g., other procedures, forms, drawings, etc.):

EPIP 1.2, Emergency Classification, EPIP 1.2.1, Emergency Action Levels TEchnical Basis

Training review recommended Per NP 1.1.3? ☐ NO ☒ YES (If Yes, RFT Number Per NP 1.1.3 CA 052693

Document Preparer (print/sign) Donna M Flanagan / D M Flanagan Date 10-11-04

Indicates draft prepared according to NP1.1.3, any commitments/bases changes have been documented and resolved.

II - TECHNICAL REVIEW

(Tech review cannot be the Preparer or Approval Authority)

Technical Reviewer (print/sign) RICHARD JOHNSON / Richard Johnson Date 10/12/04

Indicates draft technically correct, consistent with references/bases/upper tier requirements, requirements of NP 1.1.3 completed.

III – DOCUMENT OWNER REVIEW

QC review required according to NP 1.1.3/NP 8.4.1? ☒ NA ☐ YES (If yes, QC Signature) ____

Required Reviewers/Organizations: ____

Validation Required? ☐ NO ☒ YES ☐ WAIVED (Group Head Approval and Reason Required)

Reason Validation Waived: ↳ See EPIP 1.2

Continue on PBF-0026c if necessary.

Validation Waiver Approval: ____

Group Head Signature

Changes pre-screened according to NP 5.1.8? ☐ NO ☒ YES (Provide documentation according to NP 5.1.8)

Screening completed according to NP 5.1.8? ☒ NA ☐ YES (Attach copy) Safety evaluation required? ☒ NO ☐ YES

Training or briefing required? ☐ NO ☒ YES If YES, training or briefing required before issue? ☐ NO ☒ YES

☐ QR/PORC Review NOT Required (Admin or NNSR only) ☐ QR Review Required ☒ PORC Review Required (reference NP 1.6.5)

Document Owner (print/sign) MONICA RAY / Monica Ray Date 10/13/04

Indicates document is technically correct, can be performed as written, does not adversely affect personnel or nuclear safety, appropriate reviews have been performed (i.e., technical, cross-disciplinary, validation and 50.59/72.48), comments have been resolved and incorporated as appropriate, affected documents/ training/briefing have been identified and word processing completed. Document Control notified if emergent issuance required (e.g., may be less than 2 days for procedure issuance)

IV - APPROVAL

(The Preparer, Qualified Reviewer (QR), and Approval Authority shall be different individuals)

QR/PORC (print/sign) T. Skow / [illegible] Date 10/14/04

Indicates 50.59/72.48 applicability assessed, any necessary screenings/evaluations performed, determination made as to whether additional cross-disciplinary review required, and if required, performed.

PORC Meeting No. ____

Approval Authority (print/sign) MONICA RAY / Monica Ray Date 10/14/04

V – RELEASE FOR DISTRIBUTION

☐ NA ☐ YES Pre-implementation requirements complete (e.g., training/briefings, affected documents, word processing, etc.).

☐ Specific effective date not required. Issue per Document Control schedule.

☐ Required effective date: ____ (Coordinate date with Document Control)

Document Owner/Designee (print/sign) ____ / ____ Date ____

Effective Date (to be entered by Document Control): ____

References: NP 1.1.3, NP 1.1.5, NP 1.1.6
NP 1.2.3, NP 1.2.5, NP 1.2.6

10CFR 50.54(q) REVIEW FORM

Description of Change: REVISE CURRENT EAL SCHEME TO NEI 99-01 FORMAT.

☒ **Plan Sections/Procedure(s) #:** EP APPENDIX B **Revision(s) #:** 22

☐ **Mod #:**

☐ **Other:**

Is the proposed change purely editorial in nature (see definition)? *[If YES, discontinue review process and process the procedure change.]* ☐ YES ☒ NO

Does the proposed change affect any of the following: [*Check 'yes' or 'no'. Reference the actual standards/elements.*]

50.47	PARAPHRASED STANDARD	YES	NO
(b)(1)	Primary responsibilities of Site/NMC, State, County, or Tribal organizations.	☐	☒
	Responsibilities of supporting organizations.	☐	☒
	Initial staffing or augmentation	☐	☒
(b)(2)	On-shift responsibilities for emergency response.	☐	☒
	Staffing for initial accident response	☐	☒
	Timely augmentation	☐	☒
	Interfaces among onsite and offsite response activities.	☐	☒
(b)(3)	Arrangements for requesting and using assistance resources.	☐	☒
	Accommodations at the EOF for state and county staff.	☐	☒
	Other organizations capable of augmenting response are identified.	☐	☒

50.47	PARAPHRASED STANDARD	YES	NO
(b)(4) RSPS	Emergency classification and action level scheme.	☒	☐
	State/county minimum response based on site information.	☐	☒
	EAL Initiating Condition setpoints, or thresholds.	☒	☐
(b)(5) RSPS	Process for notification of state/county response organizations.	☐	☒
	Notification of emergency personnel.	☐	☒
	Procedure for initial and follow-up messages.	☐	☒
	ANS notification within the 10-mile EPZ	☐	☒
(b)(6)	Provisions for prompt communication among principal response organizations to emergency response personnel and to the public.	☐	☒
(b)(7)	Public information distributed on a periodic basis.	☐	☒
	News media points of contact established.	☐	☒
	Procedures for coordinated dissemination of info to the public.	☐	☒
(b)(8)	Emergency response facilities, equipment, and maintenance.	☐	☒
(b)(9) RSPS	Methods, systems, or equipment for assessing and monitoring actual or potential offsite consequences.	☐	☒
(b)(10) RSPS	Range of protective actions for the Plume EPZ established.	☐	☒
	Guidelines for choice of PARs in place.	☐	☒
	Protective actions for Ingestion Pathway EPZ established.	☐	☒
(b)(11)	Controlling radiological exposure for emergency workers.	☐	☒
(b)(12)	Arrangements for medical service for contaminated injured individuals.	☐	☒
(b)(13)	General plans for recovery and reentry.	☐	☒
(b)(14)	Exercise or drill conduct and corrective action system.	☐	☒
(b)(15)	Radiological emergency response training.	☐	☒

50.47	PARAPHRASED STANDARD	YES	NO
(b)(16)	Responsibilities for plan development, review, and distribution of emergency procedures established.	☐	☒
	EP Staff is properly trained.	☐	☒
EP	Implementation of other federal regulations and requirements or formal commitments related to the Emergency Preparedness Program.	☐	☒
ERDS	The operation, maintenance, or testing requirements of the ERDS.	☐	☒

App. E	PARAPHRASED ELEMENT	YES	NO
IV. A	Organization	☐	☒
IV. B	Assessment actions	☒	☐
IV. C	Activation of emergency response	☐	☒
IV. D	Notification procedures	☐	☒
IV E	Emergency facilities and equipment	☐	☒
IV. F	Training	☐	☒
IV. G	Maintaining emergency preparedness	☐	☒
IV. H	Recovery	☐	☒

STANDARDS AND/OR ELEMENTS EFFECTED	DESCRIPTION OF EFFECT	DECREASED EFFECTIVENESS? YES	DECREASED EFFECTIVENESS? NO
10CFR 50.47 (b)(4) 10CFR 50 APPENDIX E (IV) (B)	**Background and Scope:** SEE ATTACHED	☒	☐
	Program Requirements: SEE ATTACHED		
	Change Comparison: SEE ATTACHED		
	Change Assessment: SEE ATTACHED		
	Justification: SEE ATTACHED		

	YES	NO
This procedure can be processed without prior NRC approval.	☐	☒
This procedure change requires prior NRC approval.	☒	☐
Document all references used for this review:		
10CFR 50		
10CFR 50 APPENDIX E		

DESCRIPTION OF EFFECT
10CFR50.47 (b) (4)

Background and Scope:

The current Point Beach Nuclear Plant (PBNP) emergency action level classification scheme is based on joint NRC/FEMA guidelines contained in Appendix 1 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," October 1980. Based on the NRC endorsement of NEI 99-01, Revision 4 per RIS2003-18, it is the decision of Point Beach management to implement NEI 99-01 Revision 4, "Methodology for Development of Emergency Action Levels", at Point Beach Nuclear Plant. This revision to the EPIPs incorporates the NEI 99-01 classification scheme, with the exception of PERMANENTLY DEFUELED STATION INITIATING CONDITIONS, into the PBNP Emergency Plan.

Program Requirements:

10CFR 50.47 (b) (4) requires the following:

"A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures."

The proposed change will ensure maintained compliance with this requirement.

Change Comparison:

The NUREG 0654 guidelines were based on specific instruments, parameters and/or equipment status. NEI 99-01, as it will be incorporated into the PBNP emergency response, will clearly define conditions that represent increasing risk to the public and can provide more consistency in the classification process. The NEI 99-01 EALs also identify challenges to safety during shutdown and refueling operations and assign EALs at various modes of operation, which the NUREG 0654 guidelines do not. Based on the new philosophy of NEI 99-01, this will be an enhancement to the EP program.

Change Assessment:

This revision to the EPIPs and EAL scheme changes the basis for how classifications will be made at PBNP, and will result in changes to initiating condition levels. As such, this revision is a change to our original commitment to the NRC and will require prior approval by the NRC.

Justification:

The change from NUREG 0654 to NEI 99-01 enhances the ability for the site to classify events by consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur at nuclear power plants during plant shutdown conditions. This enhancement to the classification strategy at PBNP is backed by the NRC endorsement of NEI 99-01, Revision 4, per RIS2003-18.

DESCRIPTION OF EFFECT
10CFR50, Appendix E (IV) (B)

Background and Scope:

The current Point Beach Nuclear Plant (PBNP) emergency action level classification scheme is based on joint NRC/FEMA guidelines contained in Appendix 1 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," October 1980. Based on the NRC endorsement of NEI 99-01, Revision 4 per RIS2003-18, it is the decision of Point Beach management to implement NEI 99-01 Revision 4, "Methodology for Development of Emergency Action Levels", at Point Beach Nuclear Plant. This revision to the EPIPs incorporates the NEI 99-01 classification scheme, with the exception of PERMANENTLY DEFUELED STATION INITIATING CONDITIONS, into the PBNP Emergency Plan.

Program Requirements:

10CFR 50, Appendix E (IV) (B) requires the following:

"The means to be used for determining the magnitude of and for continually assessing the impact of the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. These emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by NRC. They shall also be reviewed with the State and local governmental authorities on an annual basis.."

The proposed change will ensure maintained compliance with this requirement.

Change Comparison:

The NUREG 0654 guidelines were based on specific instruments, parameters and/or equipment status. NEI 99-01, as it will be incorporated into the PBNP emergency response, will clearly define conditions that represent increasing risk to the public and can provide more consistency in the classification process. The NEI 99-01 EALs also identify challenges to safety during shutdown and refueling operations and assign EALs at various modes of operation, which the NUREG 0654 guidelines do not. Based on the new philosophy of NEI 99-01, this will be an enhancement to the EP program.

Change Assessment:

This revision to the EPIPs and EAL scheme changes the basis for how classifications will be made at PBNP, and will result in changes to initiating condition levels. As such, this revision is a change to our original commitment to the NRC and will require prior approval by the NRC.

DESCRIPTION OF EFFECT 10CFR50, Appendix E (IV) (B)
Justification: **The change from NUREG 0654 to NEI 99-01 enhances the ability for the site to classify events by consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur at nuclear power plants during plant shutdown conditions. This enhancement to the classification strategy at PBNP is backed by the NRC endorsement of NEI 99-01, Revision 4, per RIS2003-18.**

Prepared By: _____ **Date:** 10/11/04
Qualified Preparer

Reviewed By: Todd M. Tomobie **Date:** 10/13/04
Qualified Reviewer

Reviewed By: _____ **Date:** 10-13-2004
Regulatory Affairs

Approved By: Monica Ray **Date:** 10/14/04
Manager - EP

TABLE R – ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

GENERAL EMERGENCY

RG1

IC Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Modes: ALL

EAL Threshold Value (RG1.1 or RG1.2 or RG1.3)

Note: *If dose assessment results are available at the time of declaration, the classification should be based on RG1.2 instead of RG1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

RG1.1 VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Table R-4 Radiation Monitors	
Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+01 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+01 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E+00 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E+00 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E+00 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-01 µCi/cc
1 S/G Safety Valve open	2.48E-01 µCi/cc
2 S/G Safety Valves open	1.24E-01 µCi/cc
3 S/G Safety Valves open	8.25E-02 µCi/cc
4 S/G Safety Valves open	6.20E-02 µCi/cc

OR

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.

OR

RG1.3. Field survey results indicate closed window dose rates exceeding 1000 mrem/hr expected to continue for more than one hour, at or beyond site boundary.
OR
Analysis of field survey samples indicate thyroid CDE of 5000 mrem for one hour of inhalation, at or beyond site boundary.

SITE AREA EMERGENCY

RS1

IC Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mrem TEDE or 500 mrem Thyroid CDE for the Actual or Projected Duration of the Release.

Modes: ALL

EAL Threshold Value (RS1.1 or RS1.2 or RS1.3)

Note: *If dose assessment results are available at the time of declaration, the classification should be based on RS1.2 instead of RS1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

RS1.1.- VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Table R-3 Radiation Monitors	
Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+00 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+00 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E-01 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E-01 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E-01 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-02 µCi/cc
1 S/G Safety Valve open	2.48E-02 µCi/cc
2 S/G Safety Valves open	1.24E-02 µCi/cc
3 S/G Safety Valves open	8.25E-03 µCi/cc
4 S/G Safety Valves open	6.20E-03 µCi/cc

OR

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.

OR

RS1.3. Field survey results indicate closed window dose rates exceeding 100 mrem/hr expected to continue for more than one hour, at or beyond the site boundary;
OR
Analysis of field survey samples indicate thyroid CDE of 500 mrem for one hour of inhalation, at or beyond the site boundary.

ALERT

RA1

IC Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer.

Modes: ALL

EAL Threshold Value (RA1.1 or RA1.2 or RA1.3)

RA1.1. VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

OR

RA1.2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes or longer:

Table R-2 Radiation Monitors	
Monitor	Reading
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-02 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-02 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-01 µCi/cc*
2RE-309 Containment Purge Exhaust High Range	4.18E-01 µCi/cc*
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-02 µCi/cc

* with Unit 2 Containment purge or forced vent not occurring

OR

RA1.3. Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times RETS.

UNUSUAL EVENT

RU1

IC Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer.

Modes: ALL

EAL Threshold Value (RU1.1 or RU1.2 or RU1.3)

RU1.1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

OR

RU1.2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes or longer:

Table R-1 Radiation Monitors	
Monitor	Reading
RE-214 Auxiliary Building Vent Exhaust Gas	2.04E-04 µCi/cc
RE-315 Auxiliary Building Exhaust Low Range Gas	2.04E-04 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-04 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-04 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-03 µCi/cc*
2RE-309 Containment Purge Exhaust High Range Gas	4.18E-03 µCi/cc*
RE-221 Drumming Area Ventilation Gas	3.16E-04 µCi/cc
RE-325 Drumming Area Exhaust Low Range Gas	3.16E-04 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-04 µCi/cc
1(2)RE-229 Service Water Overboard	5.56E-05 µCi/cc
RE-230 Waste Water Effluent	2.06E-03 µCi/cc**

* with Unit 2 Containment purge or forced vent not occurring
** with Waste Water Effluent discharge not isolated

OR

RU1.3. Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times RETS.

TABLE R – ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
		RA2 IC **Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.** Modes: ALL EAL Threshold Value (RA2.1 or RA2.2) RA2.1. A VALID high alarm or reading on any of the following radiation monitors: o RE-105 SFP Area Low Range Radiation Monitor o RE-135 SFP Area High Range Radiation Monitor o RE 221 Drumming Area Ventilation Gas Monitor o RE 321 Drumming Area Exhaust Beta Particulate Monitor o RE 325 Drumming Area Exhaust Low Range Gas Monitor o 1(2) RE 102 El. 66' Containment Low Range Monitor o 1(2) RE-211 Containment Air Particulate Monitor o 1(2) RE 212 Containment Noble Gas Monitor OR RA2.2. Water level LESS THAN 10 ft above an irradiated fuel assembly for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering.	RU2 IC **Unexpected Rise in Plant Radiation.** Modes: ALL EAL Threshold Value (RU2.1 or RU2.2) RU2.1. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water as indicated by any of the following: • Spent fuel pool low water level alarm setpoint • Visual observation **AND** Any UNPLANNED VALID Area Radiation Monitor reading rises as indicated by: • 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 OR RU2.2. Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

TABLE R – ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
		RA3 IC **Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown** Modes: ALL EAL Threshold Value (RA3.1 or RA3.2) RA3.1. VALID radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions: Control Room (RE-101) **OR** Central Alarm Station (by survey) **OR** Secondary Alarm Station (by survey) OR RA3.2. Any VALID radiation monitor reading GREATER THAN 1 R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1). **Table H-1 Vital Areas** • 1(2) Containment Building • Primary Auxiliary Building • Turbine Building (by survey) • Control Building • Diesel Generator Building (by survey) • Gas Turbine Building (by survey) • Circ Water Pump House (by survey)	

TABLE C - COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
CG1	CS1	CA1	CU1
IC Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel. Modes: 5. Cold Shutdown 6. Refueling CG1.1. Reactor Vessel Level: 1. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Hold Up Tank (WHUT) level rise or any other indication of loss of Reactor Vessel inventory AND 2. Reactor Vessel level: a. LESS THAN [30 ft] 27 ft (RVLIS NR) or 0% indicated on LI-447/LI-447A for GREATER THAN 30 minutes OR b. cannot be monitored with indication of core uncovery for GREATER THAN 30 minutes as evidenced by any of the following: • Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr • Erratic Source Range Monitor Indication AND 3. CONTAINMENT challenged as indicated by any of the following: • GREATER THAN OR EQUAL TO 6% hydrogen concentration in containment • Containment pressure above 60 psig • CONTAINMENT CLOSURE not established	IC Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability. Modes: 5. Cold Shutdown EAL Threshold Value (CS1.1 or CS1.2) CS1.1. With CONTAINMENT CLOSURE not established: a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indication on LI-447/LI-447A OR b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise OR CS1.2. With CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indication on LI-447/LI-447A OR b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either: • Unexplained Containment Sump A or Waste Holdup Tank level rise • Erratic Source Range Monitor Indication	IC Loss of RCS Inventory Modes: 5. Cold Shutdown EAL Threshold Value (CA1.1 or CA1.2) CA1.1. Loss of RCS inventory as indicated by Reactor Vessel level LESS THAN 6% on LI-447/LI-447A OR CA1.2. Loss of RCS inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise AND RCS level cannot be monitored for GREATER THAN 15 minutes	IC RCS Leakage Modes: 5. Cold Shutdown EAL Threshold Value: (CU1.1 or CU1.2) CU1.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm. OR CU1.2. Identified leakage GREATER THAN 25 gpm

TABLE C - COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
	CS2	CA2	CU2
	IC **Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel** Modes: 6. Refueling EAL Threshold Value (CS2.1 or CS2.2) CS2.1. With CONTAINMENT CLOSURE not established: a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indicated on LI-447/LI-447A **OR** b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following: • Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr • Erratic Source Range Monitor Indication OR CS2.2. With CONTAINMENT CLOSURE established a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indicated on LI-447/LI-447A **OR** b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by one or more of the following: • Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr • Erratic Source Range Monitor Indication	**IC** **Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel.** Modes: 6. Refueling EAL Threshold Value (CA2.1 or CA2.2) CA2.1. Loss of Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 6% on LI-447/LI-447A. OR CA2.2. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise **AND** Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes	**IC** **UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel.** Modes: 6. Refueling EAL Threshold Value (CU2.1 or CU2.2) CU2.1. UNPLANNED RCS level lowering below the Reactor Vessel flange (89.1%) for GREATER THAN OR EQUAL TO 15 minutes OR CU2.2. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise **AND** Reactor Vessel level cannot be monitored

TABLE C - COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
		CA3 IC **Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.** Modes: 5.Cold Shutdown 6. Refueling Defueled **EAL Threshold Value** CA3.1. Loss of all **offsite** power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 **AND** Failure of all emergency generators to supply power to emergency busses. **AND** Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.	CU3 IC **Loss of All Offsite Power to Essential Busses for GREATER THAN 15 minutes.** Modes: 5. Cold Shutdown 6. Refueling **EAL Threshold Value** (CU3.1) CU3.1. Loss of all **offsite** power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN 15 minutes. **AND** At least 1 emergency generator is supplying power to an emergency bus.
		CA4 IC **Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Rector Vessel.** Mode: 5. Cold Shutdown 6. Refueling **EAL Threshold Value** (CA4.1 or CA4.2 or CA4.3) CA4.1. With CONTAINMENT CLOSURE and RCS integrity not established an UNPLANNED event results in RCS temperature exceeding 200 degrees F. OR CA4.2. With CONTAINMENT CLOSURE established and RCS integrity not established or RCS inventory reduced an UNPLANNED event results in RCS temperature exceeding 200 degrees F for GREATER THAN 20 minutes. OR CA4.3. An UNPLANNED event results in RCS temperature exceeding 200 degrees F for GREATER THAN 60 minutes or results in an RCS pressure rise of GREATER THAN 10 psig.	CU4 IC **UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel.** Mode: 5. Cold Shutdown 6. Refueling **EAL Threshold Value** (CU4.1 or CU4.2) CU4.1. An UNPLANNED event results in RCS temperature exceeding 200°F OR CU4.2. Loss of all RCS temperature and Reactor Vessel level indication for GREATER THAN 15 minutes.

TABLE C - COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
			CU6 IC **UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.** Modes: 5. Cold Shutdown 6. Refueling EAL Threshold Value (CU6.1 or CU6.2) CU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations. Table C-1 Onsite Communications Systems • Plant Public Address System • Security Radio • Commercial Phone System • Portable radios • Sound power phones OR CU6.2. Loss of all Table C-2 offsite communications capability. Table C-2 Offsite Communications Systems • Emergency Notification System (ENS) • Health Physics Network (HPN) • Operations Control Counterpart Link (OCCL) • Management CounterPart Link (MCL) • Protective Measures Counterpart Link (PMCL) • Reactor Safety Counterpart Link (RSCL) • Nuclear Accident Reporting System (NARS) • Commercial Phone System • General Telephone Lines • Manitowoc County Sheriff's Department Radio

TABLE C - COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
			CU7 IC **UNPLANNED Loss of Required DC Power for GREATER THAN 15 Minutes** Modes: 5. Cold Shutdown 6. Refueling **EAL Threshold Value** (CU7.1) CU7.1 UNPLANNED Loss of vital DC power to required DC busses based on bus voltage indications LESS THAN 115 VDC **AND** Failure to restore power to at least one required DC bus within 15 minutes from the time of loss. .
			CU8 IC **Inadvertent Criticality.** Modes: 5. Cold Shutdown 6. Refueling **EAL Threshold Value** CU8.1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

TABLE E – EVENTS RELATED TO ISFSI MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
			EU1 IC **Damage to a loaded cask CONFINEMENT BOUNDARY.** Mode: Not applicable EAL Threshold Value (EU1.1 or EU1.2 or EU1.3) EU1.1. Any one of the following natural phenomena events with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY. Report of plant personnel of a: Tornado strike Earthquake Flood Lightning strike OR EU1.2. Any of the following Accident conditions with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY. Vent Blockage Cask Drop Accidental Pressurization Air Vent and Outlet Shielding Reduction OR EU1.3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask CONFINEMENT BOUNDARY.
			EU2 IC ***Confirmed Security Event with potential loss of level of safety of the ISFSI.*** Mode: Not applicable EAL Threshold Value EU2.1. Security Event as determined from PBNP Physical Security Plan and reported by the Security Shift Supervisor.

TABLE F – Fission Product Barrier Degradation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier	Loss or Potential Loss of ANY two Barriers	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	ANY loss or ANY Potential Loss of Containment

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
Conditions requiring entry into Core-Cooling RED Path (CSP-C.1)	Conditions requiring entry into Core Cooling-ORANGE Path (CSP-C.2) OR Conditions requiring entry into Heat Sink-RED Path (CSP-H.1)	Not Applicable	Conditions requiring entry into RCS Integrity-RED Path (CSP-P.1) OR Conditions requiring entry into Heat Sink-RED Path (CSP-H.1)	Not Applicable	Conditions requiring entry into Containment-RED Path (CSP-Z.1)
2. Primary Coolant Activity Level		**2. RCS Leak Rate**		**2. Containment Pressure**	
Coolant Activity GREATER THAN 300 µCi/gm I-131 equivalent	Not Applicable	GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling (LESS THAN OR EQUAL TO [80 degree F] 35 degree F)	Unisolable leak exceeding 60 gpm	Rapid unexplained lowering following initial rise OR Containment pressure or sump level response not consistent with LOCA conditions	60 PSIG and rising OR Hydrogen concentration in containment GREATER THAN OR EQUAL TO 6% OR Containment pressure GREATER THAN 25 psig with LESS THAN one full train of depressurization equipment operating
3. Core Exit Thermocouple Readings		**3. SG Tube Rupture**		**3. Core Exit Thermocouple Reading**	
GREATER THAN OR EQUAL TO 1200 degree F	GREATER THAN OR EQUAL TO 700 degree F	SGTR that results in an ECCS (SI) Actuation	Not Applicable	Not applicable	Core exit thermocouples in excess of 1200 degrees F and restoration procedures not effective within 15 minutes; **OR,** core exit thermocouples in excess of 700 degrees F with reactor vessel level below 27 ft RVLIS NR (with no RCPs running) top of active fuel and restoration procedures not effective within 15 minutes

TABLE F – Fission Product Barrier Degradation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier	Loss or Potential Loss of ANY two Barriers	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	ANY loss or ANY Potential Loss of Containment

4. Reactor Vessel Water Level		**4. Containment Radiation Monitoring**		**4. SG Secondary Side Release with P-to-S Leakage**	
Not Applicable	Level LESS THAN OR EQUAL TO ▪ 25 ft RVLIS NR (with no RCPs running) • [100 ft] 90 ft RVLIS WR (with 1 RCP running) ▪ [120 ft] 110 ft RVLIS WR (with 2 RCPs running)	Containment rad monitor reading GREATER THAN 3.5 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128	Not Applicable	RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate greater than 10 gpm with nonisolable steam release from affected S/G to the environment	Not applicable
5. Containment Radiation Monitoring		**5. Other Indications**		**5. CNMT Isolation Valves Status After CNMT Isolation**	
Containment rad monitor reading GREATER THAN 17 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128	Not Applicable	Not Applicable	Not Applicable	Containment isolation valve(s) not closed AND Downstream pathway to the environment exists, after containment isolation	Not Applicable
6. Other Indications		**6. Emergency Director Judgment**		**6. Significant Radioactive Inventory in Containment**	
Failed Fuel Monitor (RE-109) reading GREATER THAN OR EQUAL TO 4500 mR/hr	Not Applicable	Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier		Not Applicable	Containment rad monitor reading GREATER THAN 15,900 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128
7. Emergency Director Judgment				**7. Other Indications**	
Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier				Not Applicable	Not Applicable
				8. Emergency Director Judgment	
				Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	

TABLE H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

GENERAL EMERGENCY HG1	SITE AREA EMERGENCY HS1	ALERT HA1	UNUSUAL EVENT HU1
IC **Security Event Resulting in Loss Of Physical Control of the Facility.** Mode: ALL **EAL Threshold Value** HG1.1. A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of **EITHER:** A VITAL AREA such that operation of equipment required for safe shutdown is lost **OR** Spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in the pool).	**IC** **Confirmed Security Event in a Plant VITAL AREA.** Mode: ALL **EAL Threshold Value** (HS1.1 or HS1.2) HS1.1. INTRUSION into the plant VITAL AREA by a HOSTILE FORCE. OR HS1.2. Security Supervision reports confirmed sabotage discovered in a VITAL AREA	**IC** **Natural and Destructive Phenomena Affecting the Plant VITAL AREA.** Mode: ALL **EAL Threshold Value** (HA1.1 or HA1.2 or HA1.3 or HA1.4 or HA1.5 or HA1.6) HA1.1. Seismic event GREATER THAN Operating Basis Earthquake (OBE) as indicated by: VALID seismic monitor indication of ground acceleration **EITHER:** GREATER THAN OR EQUAL TO 0.06 g horizontal **OR** GREATER THAN OR EQUAL TO 0.04 g vertical OR HA1.2. Tornado or high winds (15 minute average) GREATER THAN 108 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures / equipment or Control Room indication of degraded performance of those systems. • Reactor Building • Intake Building • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room OR HA1.3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or control room indication of degraded performance of those systems: • Reactor Building • Intake Building • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room OR	**IC** **Natural and Destructive Phenomena Affecting the PROTECTED AREA.** Mode: ALL **EAL Threshold Value** (HU1.1 or HU1.2 or HU1.3 or HU1.4 or HU1.5 or HU1.6 or HU1.7) HU1.1. Earthquake felt in plant as indicated by: Activation of 2 or more seismic monitors **AND** Verified by: • Actual ground shaking **OR** • By contacting the U.S. Geological Survey National Earthquake Information Center OR HU1.2. Report by plant personnel of tornado or high winds GREATER THAN 108 mph (15 minute average) striking within PROTECTED AREA boundary. OR HU1.3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary. OR HU1.4. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment. OR HU1.5. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. OR HU1.6. Uncontrolled flooding in the following areas of the plant that has the potential to affect safety related equipment needed for the current operating mode: • Auxiliary building caused by rupture of the SW header **OR** • Water intake structure caused by rupture of a circulating water system expansion joint or fire water main. OR

TABLE H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
		HA1.4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas: • Reactor Building • Intake Building • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room OR HA1.5. Uncontrolled flooding in following areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment. Auxiliary building caused by rupture of the SW header **OR** Water intake structure caused by rupture of a circulating water system expansion joint or fire water main. OR HA 1.6 Lake (forebay) level GREATER THAN OR EQUAL TO 9.0 ft (588.2 ft IGLD)	HU 1.7 Lake (forebay) level GREATER THAN OR EQUAL TO 8.0 ft (587.2 ft IGLD)

TABLE H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

GENERAL EMERGENCY

HG2

IC Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.

Mode: ALL

HG2.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

HS2

IC Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.

Mode: ALL

EAL Threshold Value

HS2.1. Control room evacuation has been initiated.

AND

Control of the plant cannot be established per AOP-10A Safe Shutdown - Local Control within 15 minutes.

HS3

IC Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.

Mode: ALL

EAL Threshold Value

HS3.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for 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

HA2

IC FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Mode: ALL

EAL Threshold Value

HA2.1. FIRE or EXPLOSION in any of the following areas (Table H-1):

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

AND

Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.

HA3

IC Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.

Mode: ALL

EAL Threshold Value (HA3.1 or HA3.2)

HA3.1. Report or detection of toxic gases within or contiguous to a VITAL AREA (Table H-1) in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

OR

UNUSUAL EVENT

HU2

IC FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.

Mode: All

EAL Threshold Value

HU2.1. FIRE in buildings or areas contiguous to any of the Table H-1 areas not extinguished within 15 minutes of control room notification or verification of a control room alarm

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

HU3

IC Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.

Mode: ALL

EAL Threshold Value (HU3.1 or HU3.2)

HU3.1. Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.

OR

HU3.2. Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.

TABLE H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
		HA3.2. Report or detection of gases in concentration GREATER THAN the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA (Table H-1). **Table H-1 VITAL AREAS** • 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House	
		HA4 **IC Confirmed Security Event in a Plant PROTECTED AREA.** Mode: ALL EAL Threshold Value (HA4.1 or HA4.2) HA4.1. INTRUSION into the PROTECTED AREA by a HOSTILE FORCE. OR HA4.2. Security Shift Supervisor reports any of the following: ▪ Sabotage device discovered in the PROTECTED AREA ▪ Standoff attack on the site protected area by a HOSTILE FORCE (i.e., Sniper) ▪ ANY Security event of increasing severity that persists for GREATER THAN 30 minutes: • Credible bomb threats • Extortion • Suspicious Fire or Explosion • Significant Security System Hardware Failure • Loss of Guard Post Contact	HU4 **IC Confirmed Security Event Which Indicates a Potential Degradation In the Level of Safety of the Plant.** Mode: ALL EAL Threshold Value (HU4.1 or HU4.2) HU4.1. Security Shift Supervisor reports ANY of the following: ▪ Suspected sabotage device discovered within the plant Protected Area ▪ Suspected sabotage device discovered outside the Protected Area or in the plant switchyard ▪ Confirmed tampering with safety-related equipment ▪ A hostage situation that disrupts NORMAL PLANT OPERATIONS ▪ Civil disturbance or strike which disrupts NORMAL PLANT OPERATIONS ▪ Internal disturbance that is not a short lived or that is not a harmless outburst involving ANY individuals within the Protected Area ▪ Malevolent use of a vehicle outside the Protected Area which disrupts NORMAL PLANT OPERATIONS OR HU4.2 Notification of a credible site-specific threat by the Security Shift Supervisor or outside agency (e.g., NRC, military or law enforcement)

TABLE H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
		HA5 **IC Control Room Evacuation Has Been Initiated.** Mode: ALL **EAL Threshold Value** HA5.1. Entry into AOP-10 Control Room Inaccessibility for control room evacuation.	HU5 **IC Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE.** Mode: ALL **EAL Threshold Value** HU5.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.
		HA6 **IC Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.** Mode: ALL **EAL Threshold Value** HA6.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	

TABLE S - SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SG1	SS1	SA1	SU1
IC **Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.** Modes: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** SG1.1. Loss of all **offsite** power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 **AND** Failure of all emergency diesel generators to supply power to safety-related 4160 VAC buses. **AND** Either of the following: (a or b) a. Restoration of at least one safety-related 4160 VAC bus within 4 hours is NOT likely **OR** b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by conditions requiring entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)	IC **Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.** Modes: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** SS1.1. Loss of all **offsite** power to safety-related 4160 VAC buses 1(2)-A05 AND 1(2)-A06 **AND** Failure of all emergency generators to supply power to safety-related 4160 VAC buses. **AND** Failure to restore power to at least one safety-related 4160 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.	Deleted	IC **Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes.** Modes: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** SU1.1. Loss of all **offsite** power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN 15 minutes. **AND** At least 1 emergency generator is supplying power to an emergency bus.
SG2	SS2	SA2	SU2
IC **Failure of Reactor Protection System to Complete an Automatic Trip and Manual Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.** Modes: 1. Power Operation 2. Startup **EAL Threshold Value:** SG2.1. Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%. **AND** Either of the following: (a or b) a. Indication(s) exists that the core cooling is extremely challenged Core Cooling - RED path (CSP-C.1). **OR** b. Indication(s) exists that heat removal is extremely challenged Heat Sink - RED path (CSP-H.1).	IC **Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was NOT Successful.** Modes: 1. Power Operation 2. Startup **EAL Threshold Value** SS2.1. Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%.	IC **Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was Successful.** Mode: 1. Power Operation 2. Startup 3. Hot Standby **EAL Threshold Value:** SA2.1. Indication(s) exist that a Reactor Protection System (RPS) setpoint was exceeded **AND** RPS automatic trip did NOT reduce power to LESS THAN 5% **AND** Any of the following operator actions are successful in reducing power to LESS THAN 5%: • Use of Reactor Trip Buttons • De-energizing 1(2)B01 and 1(2)B02	IC **Inability to Reach Required Shutdown Within Technical Specification Limits.** Mode: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** SU2.1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

TABLE S - SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
	SS3 IC **Loss of All Vital DC Power.** Modes: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** SS3.1. Loss of all vital DC power based on LESS THAN 115 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for GREATER THAN 15 minutes.		SU3 IC **UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN 15 Minutes.** Mode: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** SU3.1. UNPLANNED loss of most or all annunciators or indicators associated with the following safety systems for GREATER THAN 15 minutes • ECCS • Containment Isolation • Reactor Trip • Process or Effluent Radiation Monitors • Electrical Distribution/Diesel Generators
	SS4 IC **Complete Loss of Heat Removal Capability.** Modes: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** SS4.1. Loss of core cooling (CSP-C.1) AND heat sink (CSP-H.1).	SA4 IC **UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable.** Mode: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value:** SA4.1. UNPLANNED loss of most or all annunciators or indicators associated with the following safety systems for GREATER THAN 15 minutes • ECCS • Containment Isolation • Reactor Trip • Process or Effluent Radiation Monitors • Electrical Distribution/Diesel Generators **AND** Either of the following: (a or b) a. A SIGNIFICANT TRANSIENT is in progress. **OR** b. Compensatory non-alarming indications are unavailable.	SU4 IC **Fuel Clad Degradation.** Mode: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value (SU4.1 or SU4.2)** SU4.1. Failed Fuel Monitor (RE-109) GREATER THAN 750 mR/hr indicating fuel clad degradation OR SU4.2. Coolant sample activity GREATER THAN 50 µCi/gm dose equivalent I-131 indicating fuel clad degradation.

TABLE S - SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
		SA5 IC **AC power capability to essential busses reduced to a single power source for GREATER THAN 15 minutes such that any additional single failure would result in station blackout.** Modes: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** SA5.1. AC power capability to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 reduced to only one of the following sources for GREATER THAN 15 minutes • A single emergency diesel generator (G01, G02, G03 or G04) • LVSAT 1(2)-X04 • Cross-tying with the opposite unit power supply **AND** Any additional single failure will result in station blackout.	SU5 IC **RCS Leakage.** Modes: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** (SU5.1 or SU5.2) SU5.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm. OR SU5.2. Identified leakage GREATER THAN 25 gpm.

TABLE S - SYSTEM MALFUNCTION

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
	SS6 IC **Inability to Monitor a SIGNIFICANT TRANSIENT In Progress** Modes: Power Operation Startup Hot Standby Hot Shutdown **EAL Threshold Value:** SS6.1. Loss of most or all annunciator associated with the following safety systems • ECCS • Containment Isolation • Reactor Trip • Process or Effluent Radiation Monitors • Electrical Distribution/Diesel Generators **AND** SIGNIFICANT TRANSIENT in progress. **AND** *Compensatory non-alarming indications are* unavailable. **AND** Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, and maintain containment intact are unavailable.		SU6 IC **UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.** Modes: 1. Power Operation 2. Startup 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value** (SU6.1 or SU6.2) SU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations. **Table C-1 Onsite Communications Systems** • Plant Public Address System • Security Radio • Commercial Phone System • Portable radios • Sound power phones OR SU6.2. Loss of all Table C-2 offsite communications capability. **Table C-2 Offsite Communications Systems** • Emergency Notification System (ENS) • Health Physics Network (HPN) • Operations Control Counterpart Link (OCCL) • Management Counterpart Link (MCL) • Protective Measures Counterpart Link (PMCL) • Reactor Safety Counterpart Link (RSCL) • Nuclear Accident Reporting System (NARS) • Commercial Phone System • General Telephone Lines • Manitowoc County Sheriff's Department Radio
			SU8 IC **Inadvertent Criticality** Modes: 3. Hot Standby 4. Hot Shutdown **EAL Threshold Value:** (SU8.1) SU8.1 An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL

Note: Refer to NP 1.1.3 for requirements.

I - INITIATION

Doc Number EPIP 1.2 Unit PB0 Usage Level Reference Proposed Rev No 44

Title Emergency Classification Classification NNSR

☒ Revision ☐ Cancellation ☐ New Document ☐ Other (e.g., periodic review, admin hold) _____

List Temporary Changes/Feedbacks Incorporated: CAP050434, CAP058504

Description of Alteration/Reason (If necessary, continue description of changes on PBF-0026c and attach.)

Revising Emergency Action Level (EAL) scheme from NUREG 0654 basis to NEI 99-01 Basis

List other documents required to be effective concurrently with the revision (e.g., other procedures, forms, drawings, etc.):

Emergency Plan Appendix B, EPIP 1.2.1

Training review recommended Per NP 1.1.3? ☐ NO ☒ YES (If Yes, RFT Number Per NP 1.1.3 CA052693

Document Preparer (print/sign) Donna M Flanagan Date 10-11-04

Indicates draft prepared according to NP1.1.3, any commitments/bases changes have been documented and resolved.

II - TECHNICAL REVIEW

(Tech review cannot be the Preparer or Approval Authority)

Technical Reviewer (print/sign) RICHARD JOHNSON Date 10/12/04

Indicates draft technically correct, consistent with references/bases/upper tier requirements, requirements of NP 1.1.3 completed.

III – DOCUMENT OWNER REVIEW

QC review required according to NP 1.1.3/NP 8.4.1? ☒ NA ☐ YES (If yes, QC Signature) _____

Required Reviewers/Organizations: _____

Validation Required? ☐ NO ☒ YES ☐ WAIVED (Group Head Approval and Reason Required)

Reason Validation Waived: _____

Continue on PBF-0026c if necessary.

Validation Waiver Approval: _____

Group Head Signature

Changes pre-screened according to NP 5.1.8? ☐ NO ☒ YES (Provide documentation according to NP 5.1.8)

Screening completed according to NP 5.1.8? ☒ NA ☐ YES (Attach copy) Safety evaluation required? ☒ NO ☐ YES

Training or briefing required? ☐ NO ☒ YES If YES, training or briefing required before issue? ☐ NO ☒ YES

☐ QR/PORC Review NOT Required (Admin or NNSR only) ☐ QR Review Required ☒ PORC Review Required (reference NP 1.6.5)

Document Owner (print/sign) MONICA RAY Date 10/13/04

Indicates document is technically correct, can be performed as written, does not adversely affect personnel or nuclear safety, appropriate reviews have been performed (i.e., technical, cross-disciplinary, validation and 50.59/72.48), comments have been resolved and incorporated as appropriate, affected documents/ training/briefing have been identified and word processing completed. Document Control notified if emergent issuance required (e.g., may be less than 2 days for procedure issuance)

IV - APPROVAL

(The Preparer, Qualified Reviewer (QR), and Approval Authority shall be different individuals)

QR/PORC (print/sign) J. Skiel Date 10/14/04

Indicates 50.59/72.48 applicability assessed, any necessary screenings/evaluations performed, determination made as to whether additional cross-disciplinary review required, and if required, performed.

PORC Meeting No. _____

Approval Authority (print/sign) MONICA RAY Date 10/14/04

V – RELEASE FOR DISTRIBUTION

☐ NA ☐ YES Pre-implementation requirements complete (e.g., training/briefings, affected documents, word processing, etc.).

☐ Specific effective date not required. Issue per Document Control schedule.

☐ Required effective date: _____ (Coordinate date with Document Control)

Document Owner/Designee (print/sign) _____ Date _____

Effective Date (to be entered by Document Control): _____

References: NP 1.1.3, NP 1.1.5, NP 1.1.6
NP 1.2.3, NP 1.2.5, NP 1.2.6

10CFR 50.54(q) REVIEW FORM

Description of Change: REVISE CURRENT EAL SCHEME TO NEI 99-01 FORMAT

☒ **Plan Sections/Procedure(s) #:** EAP 1.2 **Revision(s) #:** 44

☐ **Mod #:**

☐ **Other:**

Is the proposed change purely editorial in nature (see definition)? *[If YES, discontinue review process and process the procedure change.]* ☐ YES ☒ NO

Does the proposed change affect any of the following: [*Check 'yes' or 'no'. Reference the actual standards/elements.*]

50.47	PARAPHRASED STANDARD	YES	NO
(b)(1)	Primary responsibilities of Site/NMC, State, County, or Tribal organizations.	☐	☒
	Responsibilities of supporting organizations.	☐	☒
	Initial staffing or augmentation	☐	☒
(b)(2)	On-shift responsibilities for emergency response.	☐	☒
	Staffing for initial accident response	☐	☒
	Timely augmentation	☐	☒
	Interfaces among onsite and offsite response activities.	☐	☒
(b)(3)	Arrangements for requesting and using assistance resources.	☐	☒
	Accommodations at the EOF for state and county staff.	☐	☒
	Other organizations capable of augmenting response are identified.	☐	☒

50.47	PARAPHRASED STANDARD	YES	NO
(b)(4) RSPS	Emergency classification and action level scheme.	☒	☐
	State/county minimum response based on site information.	☐	☒
	EAL Initiating Condition setpoints, or thresholds.	☒	☐
(b)(5) RSPS	Process for notification of state/county response organizations.	☐	☒
	Notification of emergency personnel.	☐	☒
	Procedure for initial and follow-up messages.	☐	☒
	ANS notification within the 10-mile EPZ	☐	☒
(b)(6)	Provisions for prompt communication among principal response organizations to emergency response personnel and to the public.	☐	☒
(b)(7)	Public information distributed on a periodic basis.	☐	☒
	News media points of contact established.	☐	☒
	Procedures for coordinated dissemination of info to the public.	☐	☒
(b)(8)	Emergency response facilities, equipment, and maintenance.	☐	☒
(b)(9) RSPS	Methods, systems, or equipment for assessing and monitoring actual or potential offsite consequences.	☐	☒
(b)(10) RSPS	Range of protective actions for the Plume EPZ established.	☐	☒
	Guidelines for choice of PARs in place.	☐	☒
	Protective actions for Ingestion Pathway EPZ established.	☐	☒
(b)(11)	Controlling radiological exposure for emergency workers.	☐	☒
(b)(12)	Arrangements for medical service for contaminated injured individuals.	☐	☒
(b)(13)	General plans for recovery and reentry.	☐	☒
(b)(14)	Exercise or drill conduct and corrective action system.	☐	☒
(b)(15)	Radiological emergency response training.	☐	☒

50.47	PARAPHRASED STANDARD	YES	NO
(b)(16)	Responsibilities for plan development, review, and distribution of emergency procedures established.	☐	☒
	EP Staff is properly trained.	☐	☒
EP	Implementation of other federal regulations and requirements or formal commitments related to the Emergency Preparedness Program.	☐	☒
ERDS	The operation, maintenance, or testing requirements of the ERDS.	☐	☒

App. E	PARAPHRASED ELEMENT	YES	NO
IV. A	Organization	☐	☒
IV. B	Assessment actions	☒	☐
IV. C	Activation of emergency response	☐	☒
IV. D	Notification procedures	☐	☒
IV E	Emergency facilities and equipment	☐	☒
IV. F	Training	☐	☒
IV. G	Maintaining emergency preparedness	☐	☒
IV. H	Recovery	☐	☒

STANDARDS AND/OR ELEMENTS EFFECTED	DESCRIPTION OF EFFECT	DECREASED EFFECTIVENESS? YES	DECREASED EFFECTIVENESS? NO
10CFR50.47 (b)(4) 10CFR 50 APPENDIX E (IV)(B)	**Background and Scope:** SEE ATTACHED	☒	☐
	Program Requirements: SEE ATTACHED		
	Change Comparison: SEE ATTACHED		
	Change Assessment: SEE ATTACHED		
	Justification: SEE ATTACHED		

	YES	NO
This procedure can be processed without prior NRC approval.	☐	☒
This procedure change requires prior NRC approval.	☒	☐
Document all references used for this review:		
10CFR 50		
10CFR 50 APPENDIX E		

DESCRIPTION OF EFFECT
10CFR50.47.(b) (4)

Background and Scope:

The current Point Beach Nuclear Plant (PBNP) emergency action level classification scheme is based on joint NRC/FEMA guidelines contained in Appendix 1 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," October 1980. Based on the NRC endorsement of NEI 99-01, Revision 4 per RIS2003-18, it is the decision of Point Beach management to implement NEI 99-01 Revision 4, "Methodology for Development of Emergency Action Levels", at Point Beach Nuclear Plant. This revision to the EPIPs incorporates the NEI 99-01 classification scheme, with the exception of PERMANENTLY DEFUELED STATION INITIATING CONDITIONS, into the PBNP Emergency Plan.

Program Requirements:

10CFR 50.47 (b) (4) requires the following:

"A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures."

The proposed change will ensure maintained compliance with this requirement.

Change Comparison:

The NUREG 0654 guidelines were based on specific instruments, parameters and/or equipment status. NEI 99-01, as it will be incorporated into the PBNP emergency response, will clearly define conditions that represent increasing risk to the public and can provide more consistency in the classification process. The NEI 99-01 EALs also identify challenges to safety during shutdown and refueling operations and assign EALs at various modes of operation, which the NUREG 0654 guidelines do not. Based on the new philosophy of NEI 99-01, this will be an enhancement to the EP program.

Change Assessment:

This revision to the EPIPs and EAL scheme changes the basis for how classifications will be made at PBNP, and will result in changes to initiating condition levels. As such, this revision is a change to our original commitment to the NRC and will require prior approval by the NRC.

Justification:

The change from NUREG 0654 to NEI 99-01 enhances the ability for the site to classify events by consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur at nuclear power plants during plant shutdown conditions. This enhancement to the classification strategy at PBNP is backed by the NRC endorsement of NEI 99-01, Revision 4, per RIS2003-18.

DESCRIPTION OF EFFECT
10CFR50, Appendix E (IV) (B)

Background and Scope:

The current Point Beach Nuclear Plant (PBNP) emergency action level classification scheme is based on joint NRC/FEMA guidelines contained in Appendix 1 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," October 1980. Based on the NRC endorsement of NEI 99-01, Revision 4 per RIS2003-18, it is the decision of Point Beach management to implement NEI 99-01 Revision 4, "Methodology for Development of Emergency Action Levels", at Point Beach Nuclear Plant. This revision to the EPIPs incorporates the NEI 99-01 classification scheme, with the exception of PERMANENTLY DEFUELED STATION INITIATING CONDITIONS, into the PBNP Emergency Plan.

Program Requirements:

10CFR 50, Appendix E (IV) (B) requires the following:

"The means to be used for determining the magnitude of and for continually assessing the impact of the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. These emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by NRC. They shall also be reviewed with the State and local governmental authorities on an annual basis.."

The proposed change will ensure maintained compliance with this requirement.

Change Comparison:

The NUREG 0654 guidelines were based on specific instruments, parameters and/or equipment status. NEI 99-01, as it will be incorporated into the PBNP emergency response, will clearly define conditions that represent increasing risk to the public and can provide more consistency in the classification process. The NEI 99-01 EALs also identify challenges to safety during shutdown and refueling operations and assign EALs at various modes of operation, which the NUREG 0654 guidelines do not. Based on the new philosophy of NEI 99-01, this will be an enhancement to the EP program.

Change Assessment:

This revision to the EPIPs and EAL scheme changes the basis for how classifications will be made at PBNP, and will result in changes to initiating condition levels. As such, this revision is a change to our original commitment to the NRC and will require prior approval by the NRC.

DESCRIPTION OF EFFECT 10CFR50, Appendix E (IV) (B)
Justification: **The change from NUREG 0654 to NEI 99-01 enhances the ability for the site to classify events by consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur at nuclear power plants during plant shutdown conditions. This enhancement to the classification strategy at PBNP is backed by the NRC endorsement of NEI 99-01, Revision 4, per RIS2003-18.**

Prepared By: _____ **Date:** 10/11/04
Qualified Preparer

Reviewed By: _____ **Date:** 10/13/04
Qualified Reviewer

Reviewed By: _____ **Date:** 10-13-2004
Regulatory Affairs

Approved By: _____ **Date:** 10/14/04
Manager - EP

EPIP 1.2

EMERGENCY CLASSIFICATION

DOCUMENT TYPE:	Technical
CLASSIFICATION:	NNSR
REVISION:	44 DRAFT
EFFECTIVE DATE:	
REVIEWER:	Plant Operation's Review Committee
APPROVAL AUTHORITY:	Department Manager
PROCEDURE OWNER (title):	Emergency Preparedness
OWNER GROUP:	Emergency Preparedness

Verified Current Copy: _____ _____ _____
Signature Date Time

List pages used for Partial Performance

_____
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Controlling Work Document Numbers

_____ _____
_____ _____
_____ _____

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1.0 PURPOSE

This procedure provides instructions to classify off-normal occurrences at PBNP into one of four standardized emergency classes.

2.0 PREREQUISITES

2.1 Responsibilities

2.1.1 This procedure is intended for immediate use by the Shift Manager (SM). Following the activation of the Emergency Operations Facility (EOF) the overall responsibility for classification is assumed by the Emergency Director (ED). The ED is supported in this effort by Control Room, TSC, and EOF personnel.

2.1.2 When relieved of Emergency Director duties by the Emergency Director, the Shift Manager shall no longer be responsible for performance of actions specified in this procedure, however as an NRC licensee the SM shall bring to the attention of the Emergency Director changing plant conditions which may affect the emergency classification.

2.1.3 Upon activation of the TSC, the Operations Coordinator shall monitor plant conditions and provide event classification recommendations to the Emergency Director.

2.1.4 Upon activation of the EOF, the EAL Monitor will monitor plant and offsite conditions and provide recommendations on changes to the Emergency Director.

2.2 Equipment

None

3.0 PRECAUTIONS AND LIMITATIONS

3.1 The notification of state and county emergency government agencies shall be initiated within 15 minutes of event classification, event termination, or change in protective action recommendations (PARS).

3.2 The notification to the NRC shall be completed immediately following state and county notifications and should not exceed 60-minutes from event classification, event termination, or change in protective action recommendations (PARS).

3.3 Certain conditions or occurrences, while not meeting the threshold for classification as an emergency, may nonetheless be reportable to the NRC per 10 CFR 50.72. (Guidance on interpretation of the 10 CFR 50.72 criteria may be found in NUREG-1022.)

3.4 Monitor plant conditions and the EALs for potential re-classification.

3.5 When Emergency conditions exist on both Units due to separate events, then each Unit should be classified separately according to the plant conditions and EALS. Units are independent of each other unless the event affects both units. If an event affects both units a single Emergency Classification is adequate.

4.0 INITIAL CONDITIONS

EPIP 1.1 has been (or had previously been) initiated by the Control Room because an off-normal occurrence exists (or has existed) at PBNP.

5.0 PROCEDURE

5.1 Classifying an Emergency

NOTE: If the EAL relates to Fission Product Barriers, EPIP 1.2.1 provides additional information on the POTENTIAL LOSS and LOSS criteria.

5.1.1 Make an initial EAL selection from Attachment A.

NOTE: A potential loss, or loss of a barrier should not be anticipated unless the trend is rapid, and the values are close to the threshold/criteria.

5.1.2 Reference the individual EAL page(s) in EPIP 1.2.1, for the EAL(s) selected. Read all fields on the page to determine/confirm that the EAL applies.

NOTE: Classifications are to be made consistent within 15 minutes once plant parameters reach an Emergency Action Level (EAL), indication in the Control Room.

5.1.3 IF an event has been categorized and the threshold of the EAL and related conditions are verified to have been met or exceeded **THEN the ED shall declare the emergency.**

a. Record this time the emergency classification, and the EAL on the NARS form Section 5.

b. IF this procedure is being implemented in the EOF, THEN make an announcement to your facility of the emergency and that you are assuming the duties of Emergency Director.

c. Return to EPIP 1.1 to ensure all appropriate actions are taken and coordinated with actions of the other ERFs if activated.

5.2 Terminating an Emergency

IF conditions have improved where an EAL is no longer met
THEN implement EPIP 12.1.

5.3 Missed Classifications

A missed classification is defined as a set of circumstances or events, which although no longer existing, if recognized at the time of their existence would have resulted in an emergency classification (i.e., met or exceeded an EAL of this procedure). Missed classifications do not include conditions described in EALs which are based on expected plant response which does not occur, but where operator action was successful- such as failure of RPS.

NOTE: **In ALL cases, the SM is vested with unilateral authority to classify an emergency and initiate any actions deemed appropriate to place the plant in a safe condition.**

5.3.1 If the missed classification would have been one classification, but current plant conditions warrant a lower classification, the lower classification shall be declared, but parties notified shall be informed of the temporary higher classification during the notification process.

5.3.2 IF NO current plant conditions meeting any EAL exist at the time of discovery of the missed classification,
THEN DO NOT declare the emergency.
However, an NRC notification should be made within one hour of the discovery of the undeclared event. Notify the Emergency Preparedness staff to ensure courtesy calls are made to offsite agencies.

5.4 Transient Events

5.4.1 For some events, the condition may be corrected before a declaration has been made. For example, an emergency classification is warranted when automatic and manual actions taken within the control room do not result in a required reactor trip. However, it is likely that actions taken outside of the control room will be successful, probably before the Emergency Director classifies the event. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses (e.g., coolant sampling, may be necessary).

In general, observe the following guidance: Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

5.4.2 There may be cases in which a plant condition that exceeded an EAL threshold was not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Rev. 2, Section 3 should be applied.

6.0 REFERENCES

6.1 Technical Specifications

6.2 Final Safety Analysis Report (FSAR) Chapter 14, Appendix A

6.3 Point Beach Nuclear Plant Emergency Plan

6.4 Point Beach Design Basis Document (DBDs)

6.5 Abnormal Operating Procedures (AOPs)

6.6 Emergency Operating Procedures (EOPs)

6.7 Emergency Contingency Actions (ECAs)

6.8 Critical Safety Procedures (CSPs)

6.9 Point Beach Setpoint Document (STPT)

6.10 Security and Safeguards Contingency Plan

6.11 WCAP 7525-L, Likelihood and Consequences of Turbine Overspeed at the Point Beach Nuclear Plant.

6.12 Reg Guide 1.115, Protection Against Low-Trajectory Turbine Missiles

6.13 EPRI Document, "Guidelines for Nuclear Plant Response to an Earthquake," dated October 1989

6.14 Probabilistic Safety Assessment - High Winds, and Others Sec 9, Rev 0, Dated July 1995

6.15 Bechtel Corporation, "Westinghouse Electric Corporation-Wisconsin Michigan Power Company-Point Beach Atomic Power Station-Design Criteria for Nuclear Power Plants Against Tornadoes," March 12, 1970, B-TOP-3.

6.16 SOER 85-5, Internal Flooding of Power Plant Buildings

6.17 NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82"

6.18 NRC Information Notice 90-08, "Kr-85 Hazards from Decayed Fuel"

6.19 NUREG-1022, Rev. 2, Event Reporting Guidelines 10CFR50.72 and 10CFR50.73.

6.20 RG 1.101, Rev 4, Emergency Planning and Preparedness for Nuclear Power Reactors

7.0 BASES

B-1 Code of Federal Regulation, 10 CFR 50

B-2 NEI 99-01 / NUMARC NESP-007, Methodology for Development of Emergency Actions Levels, Revision 4.

ATTACHMENT A
EMERGENCY ACTION LEVEL (EAL) OVERVIEW MATRIX

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal RAD Levels/Radiological Effluent — Offsite RAD Conditions	**RG1.1** 1 2 3 4 5 6 DEF VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer (Table R-4) **RG1.2** 1 2 3 4 5 6 DEF 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** 1 2 3 4 5 6 DEF Field survey results indicate closed window dose rates exceeding 1000 mrem/hr expected to continue for more than one hour, at or beyond site boundary. OR Analyses of field survey samples indicate thyroid CDE of 5000 mrem for one hour of inhalation, at or beyond site boundary Note 1: If dose assessment results are available at the time of declaration, the classification should be based on RG1.2 instead of RG1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated/completed in order to determine if the classification should be subsequently escalated.	**RS1.1** 1 2 3 4 5 6 DEF VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer. (Table R-3) **RS1.2** 1 2 3 4 5 6 DEF 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** 1 2 3 4 5 6 DEF Field survey results indicate closed window dose rates exceeding 100 mrem/hr expected to continue for more than one hour, at or beyond the site boundary; OR Analysis of field survey samples indicate thyroid CDE of 500 mrem for one hour of inhalation, at or beyond the site boundary. Note 2: If dose assessment results are available at the time of declaration, the classification should be based on RS1.2 instead of RS1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated/completed in order to determine if the classification should be subsequently escalated.	**RA1.1** 1 2 3 4 5 6 DEF VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer. **RA1.2** 1 2 3 4 5 6 DEF VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes or longer. (Table R-2) **RA1.3** 1 2 3 4 5 6 DEF Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times RETS.	**RU1.1** 1 2 3 4 5 6 DEF VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer. **RU1.2** 1 2 3 4 5 6 DEF VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes or longer. (Table R-1) **RU1.3** 1 2 3 4 5 6 DEF Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times RETS.
Abnormal RAD Levels/Radiological Effluent — Onsite RAD Conditions	NONE	NONE	**RA2.1** 1 2 3 4 5 6 DEF A VALID high alarm or reading on any of the following radiation monitors: • RE-103 SFP Area Low Range Radiation Monitor • RE-135 SFP Area High Range Radiation Monitor • RE 321 Dressing Area Ventilation Gas Monitor • RE 321 Dressing Area Exhaust Beta Particulate Monitor • RE 325 Dressing Area Exhaust Low Range Gas Monitor • 1(2) RE 102 El. 66' Containment Low Range Monitor • 1(2) RE-211 Containment Air Particulate Monitor • 1(2) RE 212 Containment Noble Gas Monitor **RA2.2** 1 2 3 4 5 6 DEF Water level LESS THAN 10 ft above an irradiated fuel assembly for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering. **RA3.1** 1 2 3 4 5 6 DEF VALID radiation monitor reading GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions: Control Room (RE 101) OR Central Alarm Station (by survey) OR Secondary Alarm Station (by survey) **RA3.2** 1 2 3 4 5 6 DEF Any VALID radiation monitor reading GREATER THAN 1 R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1).	**RU2.1** 1 2 3 4 5 6 DEF VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water as indicated by any of the following: • Spent fuel pool low water level alarm setpoint • Visual observation AND Any UNPLANNED VALID Area Radiation Monitor reading rises as indicated by: • RE-103 SFP Area Low Range Radiation Monitor • RE-135 SFP Area High Range Radiation Monitor • 1(2) RE-102 El. 66' Containment Low Range Monitor **RU2.2** 1 2 3 4 5 6 DEF Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.
Security	**HG1.1** 1 2 3 4 5 6 DEF A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of EITHER: A VITAL AREA such that operation of equipment required for safe shutdown is lost OR Spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in the pool)	**HS1.1** 1 2 3 4 5 6 DEF INTRUSION into the plant VITAL AREA by a HOSTILE FORCE. **HS1.2** 1 2 3 4 5 6 DEF Security Supervisor reports confirmed sabotage discovered in a VITAL AREA	**HA4.1** 1 2 3 4 5 6 DEF INTRUSION into the PROTECTED AREA by a HOSTILE FORCE. **HA4.2** 1 2 3 4 5 6 DEF Security Supervisor reports any of the following: • Sabotage device discovered in the PROTECTED AREA • Standoff attack on the site protected area by a HOSTILE FORCE (i.e., Sniper) • ANY Security event of increasing severity that persists for GREATER THAN 30 minutes: · Credible bomb threats · Extortion · Suspicious Fire or Explosion · Significant Security System Hardware Failure · Loss of Guard Post Control	**HU4.1** 1 2 3 4 5 6 DEF Security Shift Supervisor reports ANY of the following: • Suspected sabotage device discovered within the plant Protected Area • Suspected sabotage device discovered outside the Protected Area or in the plant switchyard • Confirmed tampering with safety-related equipment • A hostage situation that disrupts NORMAL PLANT OPERATIONS • Civil disturbance or strike which disrupts NORMAL PLANT OPERATIONS • Internal disturbance that is not a short lived or that is not a harmless outburst involving ANY individuals within the Protected Area • Malevolent use of a vehicle outside the Protected Area which disrupts NORMAL PLANT OPERATIONS **HU4.2** 1 2 3 4 5 6 DEF Notification of a credible site-specific threat by the Security Shift Supervisor or outside agency (e.g., NRC, military or law enforcement)
Fire	NONE	NONE	**HA2.1** 1 2 3 4 5 6 DEF FIRE or EXPLOSION in any of the following areas (Table H-1). AND Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area	**HU2.1** 1 2 3 4 5 6 DEF FIRE in buildings or areas contiguous to any of the Table H-1 areas not extinguished within 15 minutes of control room notification or verification of a control room alarm
Other Hazards and Failures	NONE	NONE	**HA3.1** 1 2 3 4 5 6 DEF Report or detection of toxic gases within or contiguous to a VITAL AREA (Table H-1) in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH) **HA3.2** 1 2 3 4 5 6 DEF Report or detection of gases in concentration GREATER THAN the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA (Table H-1)	**HU3.1** 1 2 3 4 5 6 DEF Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS. **HU3.2** 1 2 3 4 5 6 DEF Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.
Natural Disasters and Destructive Phenomena	NONE	NONE	**HA1.1** 1 2 3 4 5 6 DEF Seismic event GREATER THAN Operating Basis Earthquake (OBE) as indicated by: VALID seismic monitor indication of ground acceleration EITHER: GREATER THAN OR EQUAL TO 0.06 g horizontal OR GREATER THAN OR EQUAL TO 0.04 g vertical **HA1.2** 1 2 3 4 5 6 DEF Tornado or high winds (15 minute average) GREATER THAN 100 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures / equipment or Control Room indication of degraded performance of these systems • Reactor Building • Intake Building • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room **HA1.3** 1 2 3 4 5 6 DEF Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment or control room indication of degraded performance of these systems: • Reactor Building • Intake Building • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room **HA1.4** 1 2 3 4 5 6 DEF Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas. • Reactor Building • Intake Building • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room **HA1.5** 1 2 3 4 5 6 DEF Uncontrolled flooding in following areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment. Auxiliary building caused by rupture of the SW header OR Water intake structure caused by rupture of a circulating water system expansion joint or fire water main. **HA1.6** 1 2 3 4 5 6 DEF Lake (forebay) level GREATER THAN OR EQUAL TO 9.0 ft (588.2 ft IGLD)	**HU1.1** 1 2 3 4 5 6 DEF Earthquake felt in plant as indicated by: Activation of 2 or more seismic monitors AND Verified by: • Actual ground shaking OR • By contacting the U.S. Geological Survey National Earthquake Information Center **HU1.2** 1 2 3 4 5 6 DEF Report by plant personnel of tornado or high winds GREATER THAN 100 mph (15 minute average) striking within PROTECTED AREA boundary. **HU1.3** 1 2 3 4 5 6 DEF Vehicle crash into plant structures or systems within PROTECTED AREA boundary. **HU1.4** 1 2 3 4 5 6 DEF Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment **HU1.5** 1 2 3 4 5 6 DEF Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **HU1.6** 1 2 3 4 5 6 DEF Uncontrolled flooding in the following areas of the plant that has the potential to affect safety related equipment needed for the current operating mode: • Auxiliary building caused by rupture of the SW header OR • Water intake structure caused by rupture of a circulating water system expansion joint or fire water main. **HU1.7** 1 2 3 4 5 6 DEF Lake (forebay) level GREATER THAN OR EQUAL TO 8.0 ft (587.2 ft IGLD)
Control Room Evacuation	NONE	**HS2.1** 1 2 3 4 5 6 DEF Control room evacuation has been initiated. AND Control of the plant cannot be established per AOP-10A Safe Shutdown - Local Control within 15 minutes	**HA5.1** 1 2 3 4 5 6 DEF Entry into AOP-10 Control Room Inaccessibility for control room evacuation.	NONE
Emergency Director Judgment	**HG2.1** 1 2 3 4 5 6 DEF Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS3.1** 1 2 3 4 5 6 DEF Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA6.1** 1 2 3 4 5 6 DEF Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU5.1** 1 2 3 4 5 6 DEF Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.
Events Related to ISFSI	NONE	NONE	NONE	**EU1.1 Mode N/A** Any one of the following natural phenomena events with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY. Report of plant personnel of a: Tornado strike Earthquake Flood Lightning strike **EU1.2 Mode N/A** Any of the following Accident conditions with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY. Vent Blockage Cask Drop Accidental Pressurization Air Vent and Outlet Shielding Reduction **EU1.3 Mode N/A** Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask CONFINEMENT BOUNDARY. **EU2.1 Mode N/A** Security Event as determined from PBNP Physical Security Plan and reported by the Security Shift Supervisor.

HOT CONDITIONS

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction – RPS Failure	SG2.1 Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%. AND Either of the following (a or b) a. Indication(s) exists that the core cooling is extremely challenged Core Cooling - RED path (CSP-C.1) OR b. Indication(s) exists that heat removal is extremely challenged Heat Sink - RED path (CSP-H.1).	SS2.1 Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%.	SA2.1 Indication(s) exist that a Reactor Protection System (RPS) setpoint was exceeded AND RPS automatic trip did NOT reduce power to LESS THAN 5% AND Any of the following operator actions are successful in reducing power to LESS THAN 5%: · Use of Reactor Trip Buttons · De-energizing 1(2)B01 and 1(2)B02	NONE
Loss of Power	SG1.1 Loss of all offsite power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 AND Failure of all emergency diesel generators to supply power to safety-related 4160 VAC buses. AND Either of the following (a or b) a. Restoration of at least one safety-related 4160 VAC bus within 4 hours is NOT likely OR b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by conditions requiring entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)	SS1.1 Loss of all offsite power to safety-related 4160 VAC buses 1(2)-A05 AND 1(2)-A06 AND Failure of all emergency generators to supply power to safety-related 4160 VAC buses. AND Failure to restore power to at least one safety-related 4160 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power. SS3.1 Loss of all vital DC power based on LESS THAN 115 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for GREATER THAN 15 minutes.	SA5.1 AC power capability to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 reduced to only one of the following sources for GREATER THAN 15 minutes · A single emergency diesel generator (G01, G02, G03 or G04) · LVSAT 1(2)-X04 · Cross-tying with the opposite unit power supply AND *Any additional single failure will result in station blackout.*	SU1.1 Loss of all offsite power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN 15 minutes AND At least 1 emergency generator is supplying power to an emergency bus
Inoperable Safe Shutdown Conditions	NONE	SS4.1 Loss of core cooling (CSP-C.1) AND heat sink (CSP-H.1)	NONE	SU2.1 Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.
INST/COMM	NONE	SS6.1 Loss of most or all annunciators associated with the following safety systems · ECCS · Containment Isolation · Reactor Trip · Process or Effluent Radiation Monitors · Electrical Distribution/Diesel Generators AND SIGNIFICANT TRANSIENT in progress. AND Compensatory non-alarming indications are unavailable. AND Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, or maintain containment intact are unavailable.	SA4.1 UNPLANNED loss of most or all annunciators or indicators associated with the following safety systems for GREATER THAN 15 minutes · ECCS · Containment Isolation · Reactor Trip · Process or Effluent Radiation Monitors · Electrical Distribution/Diesel Generators AND Either of the following: (a or b) a. A SIGNIFICANT TRANSIENT is in progress. OR b. *Compensatory non-alarming indications are unavailable.*	SU3.1 UNPLANNED loss of most or all annunciators or indicators associated with the following safety systems for GREATER THAN 15 minutes · ECCS · Containment Isolation · Reactor Trip · Process or Effluent Radiation Monitors · Electrical Distribution/Diesel Generators SU6.1 Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations. SU6.2 Loss of all Table C-2 offsite communications capability.
Fuel Clad Degradation	NONE	NONE	NONE	SU4.1 Failed Fuel Monitor (RE-109) GREATER THAN 750 mR/hr indicating fuel clad degradation SU4.2 Coolant sample activity GREATER THAN 50 µCi/gm dose equivalent I-131 indicating fuel clad degradation.
Coolant Leakage	NONE	NONE	NONE	SU5.1 Unidentified or pressure boundary leakage GREATER THAN 10 gpm. SU5.2 Identified leakage GREATER THAN 25 gpm.
Inadvertent Criticality	NONE	NONE	NONE	SU8.1 An UNPLANNED sustained positive startup rate observed on nuclear instrumentation
Fission Product Barrier Degradation	FG1.2 Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier	FS1.1 Loss or Potential Loss of ANY two Barriers	FA1.1 ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	FU1.1 ANY loss or ANY Potential Loss of Containment

Table F-1 FISSION PRODUCT BARRIER MATRIX

Fuel Clad Barrier – Loss	Fuel Clad Barrier – Potential Loss	RCS Barrier – Loss	RCS Barrier – Potential Loss	Containment Barrier – Loss	Containment Barrier – Potential Loss
1 Critical Safety Function Status Conditions requiring entry into Core Cooling - RED path (CSP-C.1)	1 Critical Safety Function Status Conditions requiring entry into Core Cooling - ORANGE path (CSP-C.2) OR Conditions requiring entry into Heat Sink - RED path (CSP-H.1)	1. Critical Safety Function Status - Not Applicable	1 Critical Safety Function Status Conditions requiring entry into RCS Integrity - RED path (CSP-P.1) OR Conditions requiring entry into Heat Sink - RED path (CSP-H.1)	1 Critical Safety Function Status - Not Applicable	1 Critical Safety Function Status Conditions requiring entry into Containment - Red PATH (CSP-Z.1)
2 Primary Coolant Activity Level GREATER THAN 300 µCi/gm I-131 equivalent	2 Primary Coolant Activity Level - Not Applicable	2 RCS Leak Rate GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling (LESS THAN OR EQUAL TO [50 degrees F] 35 degrees F)	2 RCS Leak Rate Unusable leak exceeding 60 gpm	2 Containment Pressure Rapid unexplained lowering following initial rise OR Containment pressure or sump level response not consistent with LOCA conditions	2 Containment Pressure 60 psig and rising OR Hydrogen concentration in containment GREATER THAN OR EQUAL TO 6% OR Containment pressure GREATER THAN 25 psig with LESS THAN *one full train of* depressurization equipment operating
3 Core Exit Thermocouple Readings GREATER THAN OR EQUAL TO 1200 degrees F	3 Core Exit Thermocouple Readings GREATER THAN OR EQUAL TO 700 degrees F	3 SG Tube Rupture that results in an ECCS (SI) Actuation	3 SG Tube Rupture - Not Applicable	3 Core Exit Thermocouple Reading - Not Applicable	3 Core Exit thermocouple Reading in excess of 1200 degrees F and restoration procedures not effective within 15 minutes. OR Core exit thermocouples in excess of 700 degrees F with reactor vessel level below 27 ft RVLIS NR (with no RCPs running) top of active fuel and restoration procedures not effective within 15 minutes
4 Reactor Vessel Water Level - Not Applicable	4 Reactor Vessel Water Level LESS THAN OR EQUAL TO · 25 ft RVLIS NR (with no RCPs running) · [100 ft] 90 ft RVLIS WR (with 1 RCP running) · [120 ft] 110 ft RVLIS WR (with 2 RCPs running)	4 Containment Radiation Monitor reading GREATER THAN 3.5 R/hr indicated on any of following: · 1(2) RM-1126 · 1(2) RM-1127 · 1(2) RM-1128	4. Containment Radiation Monitoring - Not Applicable	4 SG Secondary Side Release with P-to-S Leakage RUPTURED S/G to also FAULTED outside of containment OR Primary-to-Secondary leakrate greater than 10 gpm with unmonitored steam release from affected S/G to the environment	4 SG Secondary Side Release with P-to-S Leakage - Not Applicable
5 Containment Radiation Monitoring reading GREATER THAN 17 R/hr indicated on any of the following · 1(2) RM-1126 · 1(2) RM-1127 · 1(2) RM-1128	5 Containment Radiation Monitoring - Not applicable	5 Other Indications - Not Applicable	5. Other Indications - Not Applicable	5 CNMT Isolation Valves Status After CNMT Isolation-Containment isolation valve(s) not closed AND Downstream pathway to the environment exists after containment isolation	5. CNMT Isolation Valves Status After CNMT Isolation - Not Applicable
6 Other Indications Failed Fuel Monitor (RE-109) reading GREATER THAN OR EQUAL TO 4500 mrem/hr	6 Other Indications - Not Applicable	6 Emergency Director Judgment Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier	6. Emergency Director Judgment Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier	6 Significant Radioactive Inventory in Containment - Not Applicable	6 Significant Radioactive Inventory in Containment and monitor reading GREATER THAN 15,900 R/hr indicated on any of the following · 1(2) RM-1126 · 1(2) RM-1127 · 1(2) RM-1128
7. Emergency Director Judgment Any condition in the opinion of the *Emergency Director that* indicates Loss or Potential Loss of the Fuel Clad Barrier	7 Emergency Director Judgment Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel *Clad Barrier*			7. Other Indications - Not Applicable	7 Other Indications - Not Applicable
				8 Emergency Director Judgment Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment Barrier	8 Emergency Director Judgment Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment Barrier

Table R-1 Radiation Monitors

Monitor	Reading
RE-214 Auxiliary Building Vent Exhaust Gas	2.04E-04 µCi/cc
RE-315 Auxiliary Building Exhaust Low Range Gas	2.04E-04 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-04 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-04 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-03 µCi/cc*
2RE-309 Containment Purge Exhaust High Range Gas	4.18E-03 µCi/cc*
RE-221 Drumming Area Ventilation Gas	3.16E-04 µCi/cc
RE-325 Drumming Area Exhaust Low Range Gas	3.16E-04 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-04 µCi/cc
1(2)RE-229 Service Water Overboard	5.56E-05 µCi/cc
RE-230 Waste Water Effluent	2.05E-03 µCi/cc**

* with Unit 2 Containment purge or forced vent not occurring
** with Waste Water Effluent discharge not isolated

Table R-2 Radiation Monitors

Monitor	Reading
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-02 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-02 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-01 µCi/cc*
2RE-309 Containment Purge Exhaust High Range Gas	4.18E-01 µCi/cc*
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-02 µCi/cc

* with Unit 2 Containment purge or forced vent not occurring

Table R-3 Radiation Monitors

Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+00 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+00 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E-01 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E-01 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E-01 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-02 µCi/cc
1 S/G Safety Valve open	2.48E-02 µCi/cc
2 S/G Safety Valves open	1.24E-02 µCi/cc
3 S/G Safety Valves open	8.25E-03 µCi/cc
4 S/G Safety Valves open	6.20E-03 µCi/cc

Table R-4 Radiation Monitors

Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+01 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+01 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E+00 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E+00 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E+00 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-01 µCi/cc
1 S/G Safety Valve open	2.48E-01 µCi/cc
2 S/G Safety Valves open	1.24E-01 µCi/cc
3 S/G Safety Valves open	8.25E-02 µCi/cc
4 S/G Safety Valves open	6.20E-02 µCi/cc

Table C-1 Onsite Communications Systems

- Plant Public Address System
- Security Radio
- Commercial Phone System
- Portable radios
- Sound power phones

Table C-2 Offsite Communications Systems

- Emergency Notification System (ENS)
- Health Physics Network (HPN)
- Operations Control Counterport Link (OCCL)
- Management Counterport Link (MCL)
- Protective Measures Counterport Link (PMCL)
- Reactor Safety Counterport Link (RSCL)
- Nuclear Accident Reporting System (NARS)
- Commercial Phone System
- General Telephone Lines
- Manitowoc County Sheriff's Department Radio

Table H-1 Vital Areas

- 1(2) Containment Building
- Primary Auxiliary Building
- Turbine Building (by survey)
- Control Building
- Diesel Generator Building (by survey)
- Gas Turbine Building (by survey)
- Circ Water Pump House (by survey)

HOT CONDITIONS
(RCS GREATER THAN 200°F)

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | DEF Defueled

NMC — PBNP Emergency Action Level Matrix
EPIP 1.2, Emergency Classification

Approved: _____ Manager Emergency Preparedness _____ Date

COLD CONDITIONS

Cold Shutdown/Refueling System Malfunction

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Loss of Power	NONE	NONE	**CA3.1** (3 4 DEF) Loss of all offsite power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 AND Failure of all emergency generators to supply power to emergency busses. AND Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.	**CU3.1** (3 4) Loss of all offsite power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN 15 minutes AND At least 1 emergency generator is supplying power to an emergency bus. **CU7.1** (3 4) UNPLANNED Loss of vital DC power to required DC busses based on bus voltage indications LESS THAN 115 VDC AND Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.
RCS Temp	NONE	NONE	**CA4.1** (3 4) With CONTAINMENT CLOSURE and RCS integrity not established an UNPLANNED event results in RCS temperature exceeding 200 degrees F. **CA4.2** (3 4) With CONTAINMENT CLOSURE established and RCS integrity not established or RCS inventory reduced an UNPLANNED event results in RCS temperature exceeding 200 degrees F for GREATER THAN 20 minutes **CA4.3** (3 4) An UNPLANNED event results in RCS temperature exceeding 200 degrees F for GREATER THAN 60 minutes or results in an RCS pressure rise of GREATER THAN 10 psig.	**CU4.1** (3 4) An UNPLANNED event results in RCS temperature exceeding 200° F. **CU4.2** (3 4) Loss of all RCS temperature and Reactor Vessel level indication for GREATER THAN 15 minutes.
RCS Level	**CG1.1** (3 4) Reactor Vessel Level: 1. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Hold Up Tank level rise or any other indication of loss of Reactor Vessel inventory AND 2. Reactor Vessel level: a. LESS THAN [30 ft] 27 ft (RVLIS NR) or 0% indicated on LI-447/LI-447A for GREATER THAN 30 minutes OR b. cannot be monitored with indication of core uncovery for GREATER THAN 30 minutes as evidenced by any of the following: · Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr · Erratic Source Range Monitor Indication AND 3. CONTAINMENT challenged as indicated by any of the following: · GREATER THAN OR EQUAL TO 6% hydrogen concentration in containment · Containment pressure above 60 psig · CONTAINMENT CLOSURE not established	**CS1.1** (3) With CONTAINMENT CLOSURE not established: a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indication on LI-447/LI-447A OR b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise **CS1.2** (3) With CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indication on LI-447/LI-447A OR b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either: · Unexplained Containment Sump A or Waste Holdup Tank level rise · Erratic Source Range Monitor Indication **CS2.1** (4) With CONTAINMENT CLOSURE not established a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indicated on LI-447/LI-447A OR b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following: · Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr · Erratic Source Range Monitor Indication **CS2.2** (4) With CONTAINMENT CLOSURE established a. Reactor Vessel Inventory as indicated by Reactor Vessel level 0% indicated on LI-447/LI-447A OR b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by one or more of the following: - Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr - Erratic Source Range Monitor Indication	**CA1.1** (3) Loss of RCS inventory as indicated by Reactor Vessel level LESS THAN 6% on LI-447/LI-447A **CA1.2** (3) Loss of RCS inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise AND RCS level cannot be monitored for GREATER THAN 15 minutes **CA2.1** (4) Loss of Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 6% on LI-447/LI-447A. **CA2.2** (4) Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise AND Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes	**CU2.1** (4) UNPLANNED RCS level lowering below the Reactor Vessel flange (89.1%) for GREATER THAN OR EQUAL TO 15 minutes **CU2.2** (4) Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise AND Reactor Vessel level cannot be monitored
COLAM	NONE	NONE	NONE	**CU6.1** (3) Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations. **CU6.2** (3) Loss of all Table C-2 offsite communications capability.
Coolant Leakage	NONE	NONE	NONE	**CU1.1** (3) Unidentified or pressure boundary leakage GREATER THAN 10 gpm. **CU1.2** (3) Identified leakage GREATER THAN 25 gpm.
Inadvertent Criticality	NONE	NONE	NONE	**CU8.1** (3) An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Table R-1 Radiation Monitors

Monitor	Reading
RE-214 Auxiliary Building Vent Exhaust Gas	2.04E-04 µCi/cc
RE-315 Auxiliary Building Exhaust Low Range Gas	2.04E-04 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-04 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-04 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-03 µCi/cc*
2RE-309 Containment Purge Exhaust High Range Gas	4.18E-03 µCi/cc*
RE-221 Drumming Area Ventilation Gas	3.16E-04 µCi/cc
RE-325 Drumming Area Exhaust Low Range Gas	3.16E-04 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-04 µCi/cc
1(2)RE-229 Service Water Overboard	5.56E-05 µCi/cc
RE-230 Waste Water Effluent	2.06E-03 µCi/cc**

* with Unit 2 Containment purge or forced vent not occurring
** with Waste Water Effluent discharge not isolated

Table R-2 Radiation Monitors

Monitor	Reading
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-02 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-02 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-01 µCi/cc*
2RE-309 Containment Purge Exhaust High Range Gas	4.18E-01 µCi/cc*
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-02 µCi/cc

* with Unit 2 Containment purge or forced vent not occurring

Table R-3 Radiation Monitors

Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+00 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+00 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E-01 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E-01 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E-01 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-02 µCi/cc
1 S/G Safety Valve open	2.48E-02 µCi/cc
2 S/G Safety Valves open	1.24E-02 µCi/cc
3 S/G Safety Valves open	8.25E-03 µCi/cc
4 S/G Safety Valves open	6.20E-03 µCi/cc

Table R-4 Radiation Monitors

Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+01 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+01 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E+00 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E+00 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E+00 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-01 µCi/cc
1 S/G Safety Valve open	2.48E-01 µCi/cc
2 S/G Safety Valves open	1.24E-01 µCi/cc
3 S/G Safety Valves open	8.25E-02 µCi/cc
4 S/G Safety Valves open	6.20E-02 µCi/cc

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- Control Building
- Diesel Generator Building (by survey)
- Gas Turbine Building (by survey)
- Circ Water Pump House (by survey)

COLD CONDITIONS
(RCS LESS THAN OR EQUAL TO 200°F)

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refueling | DEF Defueled

NMC — PBNP Emergency Action Level Matrix
EPIP 1.2, Emergency Classification

Approved: _____ Manager Emergency Preparedness _____ Date

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL

Note: Refer to NP 1.1.3 for requirements.

I - INITIATION

Doc Number EPIP 1.2.1 Unit PB0 Usage Level Reference Proposed Rev No 0

Title Emergency Action Levels Technical Basis Classification NNSR

☐ Revision ☐ Cancellation ☒ New Document ☐ Other (e.g., periodic review, admin hold) ____

List Temporary Changes/Feedbacks Incorporated: CAP 050434, CAP 058594

Description of Alteration/Reason (If necessary, continue description of changes on PBF-0026c and attach.)

New scheme of EALs issued - changing from NUREG 0654 to NEI 99-01 basis

List other documents required to be effective concurrently with the revision (e.g., other procedures, forms, drawings, etc.):

EP Appendix B, Emergency Classification, EPIP 1.2, Emergency Classification

Training review recommended Per NP 1.1.3? ☐ NO ☒ YES (If Yes, RFT Number Per NP 1.1.3 CA 052693

Document Preparer (print/sign) Donna M Flanagan / Donna M Flanagan Date 10-11-04

Indicates draft prepared according to NP1.1.3, any commitments/bases changes have been documented and resolved.

II - TECHNICAL REVIEW

(Tech review cannot be the Preparer or Approval Authority)

Technical Reviewer (print/sign) RICHARD JOHNSON / Richard Johnson Date 10/12/04

Indicates draft technically correct, consistent with references/bases/upper tier requirements, requirements of NP 1.1.3 completed.

III - DOCUMENT OWNER REVIEW

QC review required according to NP 1.1.3/NP 8.4.1? ☒ NA ☐ YES (If yes, QC Signature) ____

Required Reviewers/Organizations: ____

Validation Required? ☐ NO ☒ YES ☐ WAIVED (Group Head Approval and Reason Required)

Reason Validation Waived: ↳ See EPIP 1.2 Documentation

Continue on PBF-0026c if necessary.

Validation Waiver Approval: ____

Group Head Signature

Changes pre-screened according to NP 5.1.8? ☐ NO ☒ YES (Provide documentation according to NP 5.1.8)

Screening completed according to NP 5.1.8? ☒ NA ☐ YES (Attach copy) Safety evaluation required? ☒ NO ☐ YES

Training or briefing required? ☐ NO ☒ YES If YES, training or briefing required before issue? ☐ NO ☒ YES

☐ QR/PORC Review NOT Required (Admin or NNSR only) ☐ QR Review Required ☒ PORC Review Required (reference NP 1.6.5)

Document Owner (print/sign) MONICA RAY / Monica Ray Date 10/13/04

Indicates document is technically correct, can be performed as written, does not adversely affect personnel or nuclear safety, appropriate reviews have been performed (i.e., technical, cross-disciplinary, validation and 50.59/72.48), comments have been resolved and incorporated as appropriate, affected documents/ training/briefing have been identified and word processing completed. Document Control notified if emergent issuance required (e.g., may be less than 2 days for procedure issuance)

IV - APPROVAL

(The Preparer, Qualified Reviewer (QR), and Approval Authority shall be different individuals)

QR/PORC (print/sign) T. Stone / [illegible] Date 10/14/04

Indicates 50.59/72.48 applicability assessed, any necessary screenings/evaluations performed, determination made as to whether additional cross-disciplinary review required, and if required, performed.

PORC Meeting No. ____

Approval Authority (print/sign) MONICA RAY / Monica Ray Date 10/14/04

V - RELEASE FOR DISTRIBUTION

☐ NA ☐ YES Pre-implementation requirements complete (e.g., training/briefings, affected documents, word processing, etc.).

☐ Specific effective date not required. Issue per Document Control schedule.

☐ Required effective date: ____ (Coordinate date with Document Control)

Document Owner/Designee (print/sign) ____ / ____ Date ____

Effective Date (to be entered by Document Control): ____

References: NP 1.1.3, NP 1.1.5, NP 1.1.6
NP 1.2.3, NP 1.2.5, NP 1.2.6

10CFR 50.54(q) REVIEW FORM

Description of Change: REVISE CURRENT EAL SCHEME TO NEI-99-01 FORMAT.

☒ **Plan Sections/Procedure(s) #:** EIP 1.2.1 **Revision(s) #:** 0

☐ **Mod #:**

☐ **Other:**

Is the proposed change purely editorial in nature (see definition)? *[If YES, discontinue review process and process the procedure change.]* ☐ YES ☐ NO

Does the proposed change affect any of the following: [*Check 'yes' or 'no'. Reference the actual standards/elements.]*

50.47	PARAPHRASED STANDARD	YES	NO
(b)(1)	Primary responsibilities of Site/NMC, State, County, or Tribal organizations.	☐	☒
	Responsibilities of supporting organizations.	☐	☒
	Initial staffing or augmentation	☐	☒
(b)(2)	On-shift responsibilities for emergency response.	☐	☒
	Staffing for initial accident response	☐	☒
	Timely augmentation	☐	☒
	Interfaces among onsite and offsite response activities.	☐	☒
(b)(3)	Arrangements for requesting and using assistance resources.	☐	☒
	Accommodations at the EOF for state and county staff.	☐	☒
	Other organizations capable of augmenting response are identified.	☐	☒

50.47	PARAPHRASED STANDARD	YES	NO
(b)(4) RSPS	Emergency classification and action level scheme.	☒	☐
	State/county minimum response based on site information.	☐	☒
	EAL Initiating Condition setpoints, or thresholds.	☒	☐
(b)(5) RSPS	Process for notification of state/county response organizations.	☐	☒
	Notification of emergency personnel.	☐	☒
	Procedure for initial and follow-up messages.	☐	☒
	ANS notification within the 10-mile EPZ	☐	☒
(b)(6)	Provisions for prompt communication among principal response organizations to emergency response personnel and to the public.	☐	☒
(b)(7)	Public information distributed on a periodic basis.	☐	☒
	News media points of contact established.	☐	☒
	Procedures for coordinated dissemination of info to the public.	☐	☒
(b)(8)	Emergency response facilities, equipment, and maintenance.	☐	☒
(b)(9) RSPS	Methods, systems, or equipment for assessing and monitoring actual or potential offsite consequences.	☐	☒
(b)(10) RSPS	Range of protective actions for the Plume EPZ established.	☐	☒
	Guidelines for choice of PARs in place.	☐	☒
	Protective actions for Ingestion Pathway EPZ established.	☐	☒
(b)(11)	Controlling radiological exposure for emergency workers.	☐	☒
(b)(12)	Arrangements for medical service for contaminated injured individuals.	☐	☒
(b)(13)	General plans for recovery and reentry.	☐	☒
(b)(14)	Exercise or drill conduct and corrective action system.	☐	☒
(b)(15)	Radiological emergency response training.	☐	☒

50.47	PARAPHRASED STANDARD	YES	NO
(b)(16)	Responsibilities for plan development, review, and distribution of emergency procedures established.	☐	☒
	EP Staff is properly trained.	☐	☒
EP	Implementation of other federal regulations and requirements or formal commitments related to the Emergency Preparedness Program.	☐	☒
ERDS	The operation, maintenance, or testing requirements of the ERDS.	☐	☒

App. E	PARAPHRASED ELEMENT	YES	NO
IV. A	Organization	☐	☒
IV. B	Assessment actions	☒	☐
IV. C	Activation of emergency response	☐	☒
IV. D	Notification procedures	☐	☒
IV E	Emergency facilities and equipment	☐	☒
IV. F	Training	☐	☒
IV. G	Maintaining emergency preparedness	☐	☒
IV. H	Recovery	☐	☒

STANDARDS AND/OR ELEMENTS EFFECTED	DESCRIPTION OF EFFECT	DECREASED EFFECTIVENESS? YES	DECREASED EFFECTIVENESS? NO
10CFR 50.47 (b)(4) 10CFR 50 APPENDIX E (iv)(B)	**Background and Scope:** SEE ATTACHED **Program Requirements:** SEE ATTACHED **Change Comparison:** SEE ATTACHED **Change Assessment:** SEE ATTACHED **Justification:** SEE ATTACHED	☒	☐

	YES	NO
This procedure can be processed without prior NRC approval.	☐	☒
This procedure change requires prior NRC approval.	☒	☐
Document all references used for this review:		
10CFR 50		
10CFR 50 APPENDIX E		

DESCRIPTION OF EFFECT
10CFR50.47 (b) (4)

Background and Scope:

The current Point Beach Nuclear Plant (PBNP) emergency action level classification scheme is based on joint NRC/FEMA guidelines contained in Appendix 1 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," October 1980. Based on the NRC endorsement of NEI 99-01, Revision 4 per RIS2003-18, it is the decision of Point Beach management to implement NEI 99-01 Revision 4, "Methodology for Development of Emergency Action Levels", at Point Beach Nuclear Plant. This revision to the EPIPs incorporates the NEI 99-01 classification scheme, with the exception of PERMANENTLY DEFUELED STATION INITIATING CONDITIONS, into the PBNP Emergency Plan.

Program Requirements:

10CFR 50.47 (b) (4) requires the following:

"A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures."

The proposed change will ensure maintained compliance with this requirement.

Change Comparison:

The NUREG 0654 guidelines were based on specific instruments, parameters and/or equipment status. NEI 99-01, as it will be incorporated into the PBNP emergency response, will clearly define conditions that represent increasing risk to the public and can provide more consistency in the classification process. The NEI 99-01 EALs also identify challenges to safety during shutdown and refueling operations and assign EALs at various modes of operation, which the NUREG 0654 guidelines do not. Based on the new philosophy of NEI 99-01, this will be an enhancement to the EP program.

Change Assessment:

This revision to the EPIPs and EAL scheme changes the basis for how classifications will be made at PBNP, and will result in changes to initiating condition levels. As such, this revision is a change to our original commitment to the NRC and will require prior approval by the NRC.

Justification:

The change from NUREG 0654 to NEI 99-01 enhances the ability for the site to classify events by consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur at nuclear power plants during plant shutdown conditions. This enhancement to the classification strategy at PBNP is backed by the NRC endorsement of NEI 99-01, Revision 4, per RIS2003-18.

DESCRIPTION OF EFFECT
10CFR50, Appendix E (IV) (B)

Background and Scope:

The current Point Beach Nuclear Plant (PBNP) emergency action level classification scheme is based on joint NRC/FEMA guidelines contained in Appendix 1 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," October 1980. Based on the NRC endorsement of NEI 99-01, Revision 4 per RIS2003-18, it is the decision of Point Beach management to implement NEI 99-01 Revision 4, "Methodology for Development of Emergency Action Levels", at Point Beach Nuclear Plant. This revision to the EPIPs incorporates the NEI 99-01 classification scheme, with the exception of PERMANENTLY DEFUELED STATION INITIATING CONDITIONS, into the PBNP Emergency Plan.

Program Requirements:

10CFR 50, Appendix E (IV) (B) requires the following:

"The means to be used for determining the magnitude of and for continually assessing the impact of the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. These emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by NRC. They shall also be reviewed with the State and local governmental authorities on an annual basis.."

The proposed change will ensure maintained compliance with this requirement.

Change Comparison:

The NUREG 0654 guidelines were based on specific instruments, parameters and/or equipment status. NEI 99-01, as it will be incorporated into the PBNP emergency response, will clearly define conditions that represent increasing risk to the public and can provide more consistency in the classification process. The NEI 99-01 EALs also identify challenges to safety during shutdown and refueling operations and assign EALs at various modes of operation, which the NUREG 0654 guidelines do not. Based on the new philosophy of NEI 99-01, this will be an enhancement to the EP program.

Change Assessment:

This revision to the EPIPs and EAL scheme changes the basis for how classifications will be made at PBNP, and will result in changes to initiating condition levels. As such, this revision is a change to our original commitment to the NRC and will require prior approval by the NRC.

DESCRIPTION OF EFFECT 10CFR50, Appendix E (IV) (B)
Justification: **The change from NUREG 0654 to NEI 99-01 enhances the ability for the site to classify events by consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur at nuclear power plants during plant shutdown conditions. This enhancement to the classification strategy at PBNP is backed by the NRC endorsement of NEI 99-01, Revision 4, per RIS2003-18.**

Prepared By: _____ **Date:** 10/11/04
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Reviewed By: Todd McDermotte **Date:** 10/13/04
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Reviewed By: _____ **Date:** 10-13-2004
Regulatory Affairs

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EPIP 1.2.1

EMERGENCY ACTION LEVEL TECHNICAL BASIS

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RU1

Initiating Condition Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer.

Operating Mode Applicability: All

EAL:

RU1.1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes or longer:

Table R-1 Radiation Monitors

Monitor	Reading
RE-214 Auxiliary Building Vent Exhaust Gas	2.04E-04 µCi/cc
RE-315 Auxiliary Building Exhaust Low Range Gas	2.04E-04 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-04 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-04 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-03 µCi/cc*
2RE-309 Containment Purge Exhaust High Range Gas	4.18E-03 µCi/cc*
RE-221 Drumming Area Ventilation Gas	3.16E-04 µCi/cc
RE-325 Drumming Area Exhaust Low Range Gas	3.16E-04 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-04 µCi/cc
1(2)RE-229 Service Water Overboard	5.56E-05 µCi/cc
RE-230 Waste Water Effluent	2.06E-03 µCi/cc**

* with Unit 2 Containment purge or forced vent not occurring
** with Waste Water Effluent discharge not isolated

RU1.3. Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times RETS.

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for 60 minutes. The fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release.

Administrative controls are established to prevent unintentional releases and to control and monitor planned releases in the Offsite Dose Calculation Manual (ODCM) [Ref. 1]. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls. Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold for this IC.

UNPLANNED includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and

the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

The Table R-1 monitor list includes monitors on all potential release pathways.

This EAL is also intended for effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs have been determined using this methodology.

This EAL also addresses uncontrolled releases that are detected by sample analyses.

PBNP Basis Reference(s):

1. RETS – PBNP Technical Specification 5.5.1 and 5.5.4
2. Offsite Dose Calculation Manual (ODCM)
3. RMS Alarm Setpoint and Response Book (RMSASRB)
4. STPT 13.4, Radiation Monitoring System: Effluent Monitors

RU2

Initiating Condition: Unexpected Rise in Plant Radiation.

Operating Mode Applicability: All

EAL:

RU2.1. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water as indicated by any of the following:

- Spent fuel pool low water level alarm setpoint
- Visual observation

AND

Any UNPLANNED VALID Area Radiation Monitor reading rises as indicated by:

- 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

RU2.2. Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels.

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Basis:

RU2.1 addresses increased radiation levels as a result of water level decreases above the Reactor Vessel flange or events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and may represent a potential degradation in the level of safety of the plant [Ref. 1]. The low level alarm is actuated by LC-634, SFP Level Indicator at 62'-8" based on maintaining at least 6' of water on a withdrawn fuel assembly [Ref. 2].

Indications include instrumentation such as water level and local area radiation monitors [Ref. 2], and personnel (e.g., refueling crew) reports. If available, security video cameras may allow remote observation.

For refueling events where the water level drops below the Reactor Vessel flange classification would be via CU2. This event escalates to an Alert per IC RA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Matrix for events in operating modes 1-4.

RU2.2 addresses UNPLANNED rises in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. This event escalates to an Alert per IC RA3 if the increase in dose rates impedes personnel access necessary for safe operation.

PBNP Basis Reference(s):

1. RMS Alarm Response Book (RMSARB)
2. DBD-13 Spent Fuel Pool Cooling and Filtration
3. STPT 13.1 Area Monitors

RA1

Initiating Condition: Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer.

Operating Mode Applicability: All

EAL:

RA1.1. VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

RA1.2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes or longer:

Table R-2 Radiation Monitors

Monitor	Reading
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.04E-02 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.04E-02 µCi/cc
2RE-307 Containment Purge Exhaust Mid Range Gas	4.18E-01 µCi/cc*
2RE-309 Containment Purge Exhaust High Range	4.18E-01 µCi/cc*
RE-327 Drumming Area Exhaust Mid Range Gas	3.16E-02 µCi/cc

* with Unit 2 Containment purge or forced vent not occurring

RA1.3. Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times RETS.

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an 15 minutes. There are administrative controls established to prevent unintentional releases, or control and monitor intentional releases located in the Radiological Effluent Technical Specifications (RETS) and implemented in the Offsite Dose Calculation Manual (ODCM) [Ref. 1, 2]. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate. Releases should not be prorated or averaged.

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

RA1.1 addresses radioactivity releases that cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. Indexing the EAL threshold to the ODCM setpoints insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

RA1.2 addresses effluent or accident radiation monitors on non-routine release pathways. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. All of the monitors included in Table R-2 provide monitoring on non-routine effluent release pathways for which a discharge permit would not normally be prepared. The reading used for the classification threshold is two hundred times the ODCM default set point for the applicable radiation monitor.

RA1.3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in lake water systems, etc.

Due to the uncertainty associated with meteorology, emergency implementing procedures call for the timely performance of dose assessments using actual (real-time) meteorology in the event of a gaseous radioactivity release of this magnitude. The results of these assessments should be compared to the ICs RS1 and RG1 to determine if the event classification should be escalated.

PBNP Basis Reference(s):

1. RETS – PBNP Technical Specifications 5.5.1 and 5.5.4
2. Offsite Dose Calculation Manual (ODCM)
3. RMS Alarm Setpoint and Response Book (RMSASRB)
4. STPT 13.4, Radiation Monitoring System: Effluent Monitors

RA2

Initiating Condition: Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

Operating Mode Applicability: All

EAL:

RA2.1. A VALID high alarm or reading on any of the following radiation monitors:

- o RE-105 SFP Area Low Range Radiation Monitor
- o RE-135 SFP Area High Range Radiation Monitor
- o RE 221 Drumming Area Ventilation Gas Monitor
- o RE 321 Drumming Area Exhaust Beta Particulate Monitor
- o RE 325 Drumming Area Exhaust Low Range Gas Monitor
- o 1(2) RE 102 El. 66' Containment Low Range Monitor
- o 1(2) RE-211 Containment Air Particulate Monitor
- o 1(2) RE 212 Containment Noble Gas Monitor

RA2.2. Water level LESS THAN 10 ft above an irradiated fuel assembly for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering.

Basis:

This IC addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent a degradation in the level of safety of the plant.

RA2.1 addresses radiation monitor indications [Ref. 1, 2, 3] of fuel uncovery and/or fuel damage. Increased readings on ventilation monitors may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Application of this EAL requires understanding of the actual radiological conditions present in the vicinity of the monitor. VALID high alarms indicated by the radiation monitors listed in RA2.1 may be indicative of a fuel handling accident and are, therefore, appropriate for this EAL [Ref. 1, 2, 3]. High alarm setpoint values and the appropriate detailed responses to radiation monitor high alarms are provided in the PBNP RMS Alarm Set Point and Response Book (RMSASRB) [Ref.1] Information Notice No. 90-08, *"KR-85 Hazards from Decayed Fuel"* was considered in establishing radiation monitor EAL thresholds and there is no impact on this EAL.

RA2.2 is based on personnel (e.g., refueling crew) reports. There is no site-specific water level instrumentation available that could be used for entry into this EAL. Water level lowering to less than 10 feet above an irradiated fuel assembly is indicative of conditions that will result in irradiated fuel uncovering while maintaining adequate radiation shielding to protect personnel in the area [Ref.7]. This EAL does not apply to planned activities that might require an irradiated fuel assembly to be raised to a level that is less than 10 feet from the surface of the water (e.g., maintenance or repair).

Escalation, if appropriate, would occur via IC RS1 or RG1 or Emergency Director judgment.

PBNP Basis Reference(s):

1. RMS Alarm Setpoint and Response Book (RMSASRB)
2. AOP-8B Irradiated Fuel Handling Accident in Containment
3. AOP-8C Fuel Handling Accident in PAB
4. STPT 13.1 Area Monitors
5. STPT 13.2 Process Monitors
6. STP 13.4 Radiation Monitoring System: Effluent Monitors
7. DBD-05, Fuel Handling System

RA3

Initiating Condition: Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown

Operating Mode Applicability: All

EAL:

RA3.1. VALID radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:

Control Room (RE-101)

OR

Central Alarm Station (by survey)

OR

Secondary Alarm Station (by survey)

RA3.2. Any VALID radiation monitor reading GREATER THAN 1 R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1).

Table H-1 Vital Areas
• 1(2) Containment Building • Primary Auxiliary Building • Turbine Building (by survey) • Control Building • Diesel Generator Building (by survey) • Gas Turbine Building (by survey) • Circ Water Pump House (by survey)

Basis:

This IC addresses increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant.

The cause and/or magnitude of the increase in radiation levels is not a concern of this IC. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved.

For RA3.1 areas requiring continuous occupancy include the Control Room, the central alarm station (CAS) and the secondary alarm station (SAS). The CAS and SAS have no installed radiation monitoring capability [Ref. 3], therefore entry into this EAL is based on radiation surveys in these areas. 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 value is without averaging, as a 30 day duration implies an event potentially more significant than an Alert. [Ref. 2, 3]

For RA3.2 areas requiring infrequent access, a valid radiation monitor reading greater than 1 R/hr would result in additional exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits and would impede necessary access.

As used here, *impede,* includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant. Table H-1 provides the list of safe shutdown areas requiring infrequent access.

The Turbine Building, Diesel Generator Building, Gas Turbine Building and Circ Water Pump House have no installed radiation monitor capability [Ref. 3], therefore entry into the EAL is based on radiation surveys in these areas.

Areas listed in Table H-1 were selected because they are areas or contiguous to areas requiring access to maintain plant safety functions.

PBNP Basis Reference(s):

1. GDC 19
2. NUREG-0737, "Clarification of TMI Action Plan Requirements", Section III.D.3
3. RMS Alarm Setpoint and Response Book (RMSASRB)

RS1

Initiating Condition: Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mrem TEDE or 500 mrem Thyroid CDE for the Actual or Projected Duration of the Release.

Operating Mode Applicability: All

Note: If dose assessment results are available at the time of declaration, the classification should be based on RS1.2 instead of RS1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

EAL:

RS1.1. VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Table R-3 Radiation Monitors

Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+00 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+00 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E-01 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E-01 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E-01 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-02 µCi/cc
1 S/G Safety Valve open	2.48E-02 µCi/cc
2 S/G Safety Valves open	1.24E-02 µCi/cc
3 S/G Safety Valves open	8.25E-03 µCi/cc
4 S/G Safety Valves open	6.20E-03 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 closed window dose rates exceeding 100 mrem/hr expected to continue for more than one hour, at or beyond the site boundary;

OR

Analysis of field survey samples indicate thyroid CDE of 500 mrem for one hour of inhalation, at or beyond the site boundary.

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed a small fraction of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

The 100 mrem TEDE dose is set at 10% of the EPA PAG, while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The Table R-3 monitor list in RS1.1 includes monitors on all potential gaseous effluent release pathways. These monitors were selected because the detectors have an operating range that enables them to provide indication for the classification threshold.

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. The monitor reading thresholds for RS1.1 were determined by multiplying the monitor readings in EAL RG1.1 Table R-4 by 0.1 to determine the monitor reading thresholds corresponding to 500 mrem Thyroid CDE at or beyond the site boundary (1 mile downwind), which is the more limiting dose value for these calculations [Ref.1]. The inputs used for the calculations in Reference 1 are described in the Basis section for EAL RG1.

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

PBNP Basis Reference(s):

1. Effluent Monitor Classification Thresholds WEDAP Basis Document, NPC 2004-00731
2. EPIP 1.3 Dose Assessment and Protective Action Recommendations
3. FSAR Table 2.6-3 Stability Index Distribution
4. FSAR Table 2.6-4 Site Atmospheric Stability Analysis Annual Average 13 month Data
5. FSAR Figure 2.6-2 Stability Class Distribution in Percent of Total Observed
6. USEPA 400-R-92-001, Manual of Protective Action Guidelines and Protective Actions for Nuclear Incidents
7. DBD-T-46 Section 3.1, Station Blackout

RG1

Initiating Condition: Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability: All

Note: If dose assessment results are available at the time of declaration, the classification should be based on RG1.2 instead of RG1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

EAL:

RG1.1. VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer:

Table R-4 Radiation Monitors

Monitor	Reading
1(2) RE-307 Containment Purge Exhaust Mid Range Gas	1.44E+01 µCi/cc
1(2) RE-309 Containment Purge Exhaust High Range Gas	1.44E+01 µCi/cc
RE-317 Auxiliary Building Exhaust Mid Range Gas	2.63E+00 µCi/cc
RE-319 Auxiliary Building Exhaust High Range Gas	2.63E+00 µCi/cc
RE-327 Drumming Area Exhaust Mid Range Gas	4.32E+00 µCi/cc
1(2) RE-231 Steam Line 1A(2A), 1(2) RE-232 Steam Line 1B(2B)	
1 Atmospheric Steam Dump Valve open	6.59E-01 µCi/cc
1 S/G Safety Valve open	2.48E-01 µCi/cc
2 S/G Safety Valves open	1.24E-01 µCi/cc
3 S/G Safety Valves open	8.25E-02 µCi/cc
4 S/G Safety Valves open	6.20E-02 µCi/cc

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 closed window dose rates exceeding 1000 mrem/hr expected to continue for more than one hour, at or beyond site boundary.

OR

Analysis of field survey samples indicate thyroid CDE of 5000 mrem for one hour of inhalation, at or beyond site boundary.

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The Table R-4 monitor list in RG1.1 includes monitors on all potential gaseous effluent release pathways. These monitors were selected because the detectors have a high enough operating range that enables them to provide indication for the classification threshold.

The monitor reading thresholds for RG1.1 were determined using the Wisconsin Electric Dose Assessment Program (WEDAP) computer code and annual average meteorology [Ref.1]. The monitor readings determined in Reference 1 were back calculated from a dose value of 5000 mrem Thyroid CDE at or beyond the site boundary (1 mile downwind), which is the more limiting dose value for these calculations.

RG1.2 dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made, the dose assessment results override the monitor reading EALs.

PBNP Basis Reference(s):

1. Effluent Monitor Classification Thresholds WEDAP Basis Document, NPC 2004-00731
2. EPIP 1.3 Dose Assessment and Protective Action Recommendations
3. FSAR Table 2.6-3 Stability Index Distribution
4. FSAR Table 2.6-4 Site Atmospheric Stability Analysis Annual Average 13 month Data
5. FSAR Figure 2.6-2 Stability Class Distribution in Percent of Total Observed
6. UDSEPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
7. DBD-T-46 Section 3.1, Station Blackout
8. FSAR Appendix A-1, Station Blackout

CU1

Initiating Condition: RCS Leakage.

Operating Mode Applicability: 5-Cold Shutdown

EAL:

CU1.1.	Unidentified or pressure boundary leakage GREATER THAN 10 gpm.
CU1.2.	Identified leakage GREATER THAN 25 gpm.

Basis:

RCS leakage at these values is considered to be a potential degradation of the level of safety of the plant in that the values indicated are well beyond Technical specification limits.

The 10 gpm threshold value for the unidentified and pressure boundary leakage was selected as it is sufficiently large to be observable via normally installed instrumentation (e.g., Pressurizer level, RCS loop level instrumentation, etc...) or reduced inventory instrumentation such as level hose indication. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). OI-55 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting.

The 25 gpm threshold value for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Prolonged loss of RCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel).

PBNP Basis Reference(s):

1. TS 3.4.13, RCS Operational Leakage limits
2. OI 55, Primary Leak Rate Calculation
3. OM 3.19, Reactor Coolant System Leakage Determination

CU2

Initiating Condition: UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: 6-Refueling

EAL:

CU2.1.	UNPLANNED RCS level lowering below the Reactor Vessel flange (89.1%) for GREATER THAN OR EQUAL TO 15 minutes
CU2.2.	Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise **AND** Reactor Vessel level cannot be monitored

Basis:

This IC is included as an UE because it may be a precursor of more serious conditions and, is considered to be a potential degradation of the level of safety of the plant. Refueling evolutions that decrease RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled.

An UNPLANNED event that results in water level decreasing below the Reactor Vessel flange warrants declaration of an Unusual Event due to the reduced RCS inventory that is available to keep the core covered. The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame.

In the refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing Containment Sump A and Waste Holdup Tank level changes [Ref. 1, 2].

Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Continued loss of RCS Inventory will result in escalation to the Alert level via either IC CA2 (Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel).

PBNP Basis Reference(s):

1. ARB C01 B 1-4, UNIT 1 CONTAINMENT SUMP A LEVEL HIGH
2. STPT 12.1, Waste Disposal System
3. OP 4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
4. OI 55, Primary Leak Rate Calculation

CU3

Initiating Condition: Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes.

Operating Mode Applicability: 5-Cold Shutdown
6-Refueling

EAL:

CU3.1.	Loss of all **offsite** power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN 15 minutes. **AND** At least 1 emergency generator is supplying power to an emergency bus.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

For the purpose of classification under this EAL, offsite power sources include any of the following:

- 345 KVAC system supplying power to the 13.8 KVAC system and the unit LVSATs
- Cross-tying with the opposite unit power supply
- Backfeeding power to the UATs through the 19 KVAC system and the main step-up transformer X-01 (X-02).

The capability to cross-tie AC power from the other unit takes credit for the redundant power source for this EAL. However, the inability to effect the cross-tie within 15 minutes warrants declaring a UE.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
2. DBD-18, 13.8 KVAC System, Section 3.3.0
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A U1, Main Power Transformer Backfeed

CU4

Initiating Condition: UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: 5-Cold Shutdown
6-Refueling

EAL:

CU4.1.	An UNPLANNED event results in RCS temperature exceeding 200°F
CU4.2.	Loss of all RCS temperature <u>and</u> Reactor Vessel level indication for GREATER THAN 15 minutes.

Basis:

This IC is included as an UE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered. In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode.

During refueling the level in the Reactor Vessel will normally be maintained above the Reactor Vessel flange. Refueling evolutions that decrease water level below the Reactor Vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/Reactor Vessel temperatures depending on the time since shutdown. Escalation to the Alert level is via CA4.

Unlike the cold shutdown mode, normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, CU4.2 would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via CA2 based on an inventory loss or CA4 based on exceeding its temperature criteria (200°F) [Ref. 1].

PBNP Basis Reference(s):

1. Technical Specifications Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
2. DBD-9, Reactor Coolant System , Rev. 3, Sections 3.23.1 to 3.23.4, 3.24.1, 3.24.2, 3.25.1 and 3.25.2
3. OI 105, RECS Heatup/Cooldown Plotting, Rev. 9
4. OP4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1, Rev. 14

CU6

Initiating Condition: UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: 5-Cold Shutdown
6-Refueling

EAL:

CU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Plant Public Address System
• Security Radio
• Commercial Phone System
• Portable radios
• Sound power phones

CU6.2. Loss of all Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• Emergency Notification System (ENS)
• Health Physics Network (HPN)
• Operations Control Counterpart Link (OCCL)
• Management Counterpart Link (MCL)
• Protective Measures Counterpart Link (PMCL)
• Reactor Safety Counterpart Link (RSCL)
• Nuclear Accident Reporting System (NARS)
• Commercial Phone System
• General Telephone Lines
• Manitowoc County Sheriff's Department Radio

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means are being utilized to make communications possible.

Table C-1 onsite communications loss [Ref. 1, 2] encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system and radios / walkie talkies).

Table C-2 offsite communications loss [Ref. 1, 2] encompasses the loss of all means of communications with offsite authorities

PBNP Basis Reference(s):

1. EPMP 2.1, Testing of Communications Equipment
2. EPMP 2.1A, Monthly Communications Test

CU7

Initiating Condition: UNPLANNED Loss of Required DC Power for Greater than 15 Minutes.

Operating Mode Applicability 5-Cold Shutdown
6-Refueling

EAL:

CU7.1	UNPLANNED Loss of vital DC power to required DC busses based on bus voltage indications LESS THAN 115 VDC **AND** Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.

Basis:

The purpose of this IC and EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

UNPLANNED is included to preclude the declaration of an emergency as a result of planned maintenance activities. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be via CA4.

LESS THAN 115 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment [Ref. 1, 2]. The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses.

PBNP Basis Reference(s):

1. FSAR Section 8.7
2. 0-SOP-DC-001/2/3/4 125 VDC System, Bus D-01, D-02, D-03, D-04 and Components

CU8

Initiating Condition: Inadvertent Criticality.

Operating Mode Applicability: 5-Cold Shutdown
6-Refueling

EAL:

CU8.1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This IC addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel mis-loading events and inadvertent dilution events. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated) which are addressed in SU8.

This condition can be identified using startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Escalation would be by Emergency Director Judgment.

PBNP Basis Reference(s):

1. OP 1B, Reactor Startup, Step 5.1 and 5.18.15

CA1

Initiating Condition: Loss of RCS Inventory.

Operating Mode Applicability: 5-Cold Shutdown

EAL:

CA1.1.	Loss of RCS inventory as indicated by Reactor Vessel level LESS THAN 6% on LI-447/LI-447A
CA1.2.	Loss of RCS inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise **AND** RCS level cannot be monitored for GREATER THAN 15 minutes

Basis:

These EALs serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel level decrease and potential core uncovery.

For CA1.1 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. The 6 inch level of the RCS Loop was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred [Ref. 1]. The 6 inch level is above the CS2 setpoint of 0 inches, the level equal to the bottom of the Reactor Vessel loop penetration. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

CA1.2 is applicable if all level indication were to be unavailable during a loss of RCS inventory event. The operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage [Ref. 1, 2].

The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 duration. The 15-minute duration allows CA1 to be an effective precursor to CS1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS1 basis. Therefore this EAL meets the definition for an Alert emergency.

If Reactor Vessel level continues to decrease then escalation to Site Area will be via CS1 (Loss of Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel).

PBNP Basis Reference(s):

1. OP 4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
2. OM 3.19, REACTOR COOLANT SYSTEM LEAKAGE DETERMINATION

CA2

Initiating Condition: Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: 6-Refueling

EAL:

CA2.1.	Loss of Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 6% on LI-447/LI-447A.
CA2.2.	Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise **AND** Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes

Basis:

These EALs serve as precursors to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel level decrease and potential core uncovery.

For CA2.1 the LI-447/LI-447A threshold corresponds to the 6 inches above the bottom inside diameter of the RCS loop [Ref. 1]. The 6 inch level of the RCS Loop was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. This condition will result in a minimum classification of Alert. The 6 inch level is above the CS2 setpoint of 0 inches, the level equal to the bottom of the Reactor Vessel loop penetration. The Bottom ID of the RCS Loop Setpoint is the level equal to the bottom of the Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

If all level indication were to be unavailable during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes [Ref. 1, 2]. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS2 duration. The 15-minute duration allows CA2 to be an effective precursor to CS2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS2 basis. Therefore this EAL meets the definition for an Alert.

If Reactor Vessel level continues to decrease then escalation to Site Area will be via CS1 (Loss of Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel).

PBNP Basis Reference(s):

1. OP 4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
2. OM 3.19, REACTOR COOLANT SYSTEM LEAKAGE DETERMINATION

CA3

Initiating Condition: Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability: 5-Cold Shutdown
6-Refueling
Defueled

EAL:

CA3.1. Loss of all **offsite** power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06

AND

Failure of all emergency generators to supply power to emergency busses.

AND

Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink.

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

This EAL is indicated by the loss of all offsite and onsite AC power to the safety-related 4160 VAC buses. Offsite power sources include 345 KVAC through the 13.8 KVAC system to the LVSAT and 345 KVAC backfed through the 19 KVAC system to the UAT. Note that the time required to effect a backfeed to the UAT is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source. Onsite power sources consist of the emergency diesel generators, the gas turbine generator, the unit main turbine generator, and power supplied from the opposite unit [Ref. 1, 2, 3, 4, 5].

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 essential busses. Even though an essential bus may be energized, if necessary loads are not operable on the energized bus then the bus should not be considered operable.

Escalating to Site Area Emergency IC SS1, if appropriate, is by Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
2. DBD-18, 13.8 KVAC System, Section 3.3.0
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A U1 Main Power Transformer Backfeed

CA4

Initiating Condition: Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: 5-Cold Shutdown
6-Refueling

EAL:

CA4.1.	With CONTAINMENT CLOSURE and RCS integrity not established an UNPLANNED event results in RCS temperature exceeding 200 degrees F.
CA4.2.	With CONTAINMENT CLOSURE established and RCS integrity not established or RCS inventory reduced an UNPLANNED event results in RCS temperature exceeding 200 degrees F for GREATER THAN 20 minutes[1].
CA4.3.	An UNPLANNED event results in RCS temperature exceeding 200 degrees F for GREATER THAN 60 minutes[1] or results in an RCS pressure rise of GREATER THAN 10 psig.

Basis:

CA4.1 addresses complete loss of functions required for core cooling during refueling and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established. No delay time is allowed for CA4.1 because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

CA4.2 addresses the complete loss of functions required for core cooling for GREATER THAN 20 minutes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is reduced (e.g., mid loop operation). The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible and is conservative given that a low pressure Containment barrier to fission product release is established.

CA4.3 addresses complete loss of functions required for core cooling for GREATER THAN 60 minutes during refueling and cold shutdown modes when RCS integrity is established. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The 10 psig pressure rise covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes.

RCS integrity is in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation.

Escalation to Site Area would be via CS1 or CS2 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

[1]Note: if an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.

PBNP Basis Reference(s):

1. Technical Specifications Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
2. OP 1A, Cold Shutdown to Hot Standby
3. Technical Specifications B 3.6.1, Containment, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
4. CL 1E, Containment Closure Checklist
5. OP 4F, Reactor Coolant System Reduced Inventory Requirements
6. DBD-09, Reactor Coolant System

CS1

Initiating Condition: Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability.

Operating Mode Applicability: 5-Cold Shutdown

EAL:

CS1.1. With CONTAINMENT CLOSURE not established:

a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indication on LI-447/LI-447A

OR

b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Holdup Tank level rise

CS1.2. With CONTAINMENT CLOSURE established:

a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indication on LI-447/LI-447A

OR

b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either:

- Unexplained Containment Sump A or Waste Holdup Tank level rise
- Erratic Source Range Monitor Indication

Basis:

Under the conditions specified by this IC, continued decrease in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to an Reactor Vessel breach, pressure boundary leakage, or continued boiling in the Reactor Vessel.

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30-minutes was chosen. The 30-minute duration allows sufficient time for actions to be performed to recover needed cooling equipment

The level associated with or without CONTAINMENT CLOSURE corresponds to the bottom inside diameter of the RCS loop. Effluent release is not expected with closure established.

If all level indication were to be unavailable during a loss of RCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Escalation to a General Emergency is via CG1 or RG1.

PBNP Basis Reference(s):

1. PBNP FSAR 4.0 Reactor Coolant System Design Basis
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. OI 55, Primary Leak Rate Calculation

CS2

Initiating Condition: Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability

Operating Mode Applicability: 6-Refueling

EAL:

CS2.1. With CONTAINMENT CLOSURE not established:

a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indicated on LI-447/LI-447A

OR

b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following:
   - Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr
   - Erratic Source Range Monitor Indication

CS2.2. With CONTAINMENT CLOSURE established

a. Reactor Vessel inventory as indicated by Reactor Vessel level 0% indicated on LI-447/LI-447A

OR

b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by one or more of the following:
   - Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr
   - Erratic Source Range Monitor Indication

Basis:

Under the conditions specified by this IC, continued decrease in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to a Reactor Vessel breach or continued boiling in the Reactor Vessel.

The level associated with or without CONTAINMENT CLOSURE corresponds to the bottom inside diameter of the RCS loop.

If all means of level monitoring are not available, the Reactor Vessel inventory loss may be detected by other indirect methods. As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment High Range Monitor indication and possible alarm. The 10 R/hr reading has been selected to be well above that expected under normal plant conditions [Ref. 1]. Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (NIS N-31 and N-32) can be used as a tool for making such determinations.

For CS2.2 in the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally available (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. Effluent release is not expected with closure established.

Escalation to a General Emergency is via CG1or RG1.

PBNP Basis Reference(s):

1. Eng Eval 2001-28, Containment High Radiation Channel Check Tolerance, 10/5/01
2. Volian Enterprises Calculation WEP-SPT-25
3. PBNP FSAR 4.0 Reactor Coolant System Design Basis, Table 4.1-6
4. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2

CG1

Initiating Condition: Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: 5-Cold Shutdown
6-Refueling

EAL:

CG1.1. Reactor Vessel Level:

1. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump A or Waste Hold Up Tank level rise or any other indication of loss of Reactor Vessel inventory

AND

2. Reactor Vessel level:

   a. LESS THAN [30 ft] 27 ft (RVLIS NR) or 0% indicated on LI-447/LI-447A for GREATER THAN 30 minutes

   OR

   b. cannot be monitored with indication of core uncovery for GREATER THAN 30 minutes as evidenced by any of the following:

   - Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr
   - Erratic Source Range Monitor Indication

AND

3. CONTAINMENT challenged as indicated by any of the following:

   - GREATER THAN OR EQUAL TO 6% hydrogen concentration in containment
   - Containment pressure above 60 psig
   - CONTAINMENT CLOSURE not established

Basis:

This EAL represents the inability to restore and maintain Reactor Vessel level to above the top of active fuel. Fuel damage is probable if Reactor Vessel level cannot be restored, as available decay heat will cause boiling, further reducing the Reactor Vessel level. Setpoints enclosed in brackets (e.g., [30 ft], etc.) are used under adverse containment conditions [Ref. 3, 4].

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

If all level indication were to be lost, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 10 R/hr setpoint has been selected to be well above that expected under normal plant conditions [Ref. 1, 2]..

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (NIS N-31 and N-32) can be used as a tool for making such determinations.

With the CONTAINMENT breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment.

When hydrogen and oxygen concentrations reach or exceed the deflagration limits (equal to or greater than 6% hydrogen), loss of the containment barrier is possible [Ref. 4, 5, 6].

The containment design pressure (60 psig) is well in excess of that expected from the design basis loss of coolant accident [Ref. 7, 8].

PBNP Basis Reference(s):

5. Eng Eval 2001-28, Containment High Radiation Channel Check Tolerance, 10/5/01
6. Volian Enterprises Calculation WEP-SPT-25
7. CSP-C.1 UNIT 1 RED, CRITICAL SAFETY PROCEDURE SAFETY RELATED RESPONSE TO INADEQUATE CORE COOLING, Step 11
8. BG-CSP-Z.1, Response to High Containment Pressure
9. EPIP 10.3, POST-ACCIDENT CONTAINMENT HYDROGEN REDUCTION
10. FSAR Section 5.1, Containment System Structure
11. BG-CSP-ST.0, CSFST

EU1

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: Not applicable

EAL:

EU1.1. Any one of the following natural phenomena events with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY:

Report of plant personnel of a:

- Tornado strike
- Earthquake
- Flood
- Lightning strike

EU1.2. Any of the following Accident conditions with resultant visible damage to or loss of a loaded cask CONFINEMENT BOUNDARY.

- Vent Blockage
- Cask Drop
- Accidental Pressurization
- Air Vent and Outlet Shielding Reduction

EU1.3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask CONFINEMENT BOUNDARY.

Basis:

A UE in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage. Table E-1 lists the criteria that define possible failure of the casl CONFINEMENT BOUNDARY.

For EU1.3, any condition not explicitly detailed as an EAL threshold value, which, in the judgment of the Emergency Director, is a potential degradation in the level of safety of the ISFSI. Emergency Director judgment is to be based on known conditions and the expected response to mitigating activities within a short time period.

PBNP Basis Reference(s):

1. VSC-24, Conditions for Cask Use and Technical Specifications Docket No. 72-1007 Certificate of Compliance No. 1007
2. Standardized NUHOMS Horizontal Modular Storage System Certificate of Compliance No. 1004
3. Technical Specification 1.2.4 of the VSC-24 Certificate of Compliance
4. AOP-8G Ventilated Storage Cask (VSC) Drop or Tipover
5. Technical Specification 1.2.7 of the NUHOMS Certificate of Compliance
6. NUHOMS SAR
7. NUH-003 Rev 8, June 2004 "FSAR for Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel

EU2

Initiating Condition: Confirmed Security Event with potential loss of level of safety of the ISFSI.

Operating Mode Applicability: Not applicable

EAL:

EU2.1.	Security Event as determined from PBNP Physical Security Plan and reported by the Security Shift Supervisor.

Basis:

This EAL is based on PBNP Physical Security Plan Security events which do not represent a potential degradation in the level of safety of the ISFSI, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72.

Reference is made to the Security Shift Supervisor because these individuals are the designated personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Security Plan.

PBNP Basis Reference(s):

1. PBNP Physical Security Plan

HU1

Initiating Condition: Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: All

EAL:

HU1.1. Earthquake felt in plant as indicated by:
Activation of 2 or more seismic monitors
AND
Verified by:

- Actual ground shaking

  OR

- By contacting the U.S. Geological Survey National Earthquake Information Center

HU1.2. Report by plant personnel of tornado or high winds GREATER THAN 108 mph striking within PROTECTED AREA boundary.

HU1.3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary.

HU1.4. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

HU1.5. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

HU1.6. Uncontrolled flooding in the following areas of the plant that has the potential to affect safety related equipment needed for the current operating mode:

- Auxiliary building caused by rupture of the SW header

  OR

- Water intake structure caused by rupture of a circulating water system expansion joint or fire water main.

HU1.7. Lake (forebay) level GREATER THAN OR EQUAL TO 8.0 ft (587.2 ft IGLD)

Basis:

UE in this IC are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the EALs define the location of the event based on the potential for damage to equipment contained therein. Escalation of the event to an Alert occurs when the magnitude of the event is sufficient to result in damage to equipment contained in the specified location.

HU1.1. Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. Method of detection is based on instrumentation, validated by a reliable source (U.S. Geological Survey National Earthquake Information Center), or operator assessment [Ref. 1, 2, 3, 4].

HU1.2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind site specific value is based on the FSAR design basis sustained wind speed of 108 mph. Wind loads of this magnitude can cause damage to safety functions. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert. [Ref. 5, 6, 7]

HU1.3 addresses crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. If the crash is confirmed to affect a plant VITAL AREA, the event may be escalated to Alert. [Ref. 7]

HU1.4 addresses only those EXPLOSIONs of sufficient force to damage permanent structures or equipment within the PROTECTED AREA. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The Emergency director also needs to consider any security aspects of the EXPLOSION, if applicable. [Ref. 7]

HU1.5 is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIREs and flammable gas build up are appropriately classified via HU2 and HU3.

HU1.6 addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The auxiliary building and water intake structure are the vulnerable areas indicated in the IPE that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged. Escalation of the emergency classification is based on the damage caused or by access restrictions that prevent necessary plant operations or systems monitoring. [Ref. 11, 12, 13, 14]

HU1.7 covers other site-specific phenomena such as flood, or seiche. This EAL can also be precursors of more serious events. Lake water level GREATER THAN OR EQUAL TO 8 feet (587.2 ft elevation IGLD) corresponds to the Turbine Building floor elevation. Both the Turbine Building and Circ Water Pumphouse are specifically susceptible to external flooding.

This EAL is consistent with the definition of a UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

PBNP Basis Reference(s):

1. AOP-28 Seismic Event
2. FSAR Volume 1 Section 2.9 Seismology
3. STPT 22.1 Seismic Event Monitoring
4. EPRI document, "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989
5. FSAR Volume 3 Page 5.1-37 Wind and Tornado Forces
6. PSA Section 9 Notebook 9.1
7. Bechtel Drawing C-3 Plant Areas
8. ARP 1(2) C33 1-2 Hydrogen Pressure High-Low Alarm
9. ARP 1(2) C33 1-3 Hydrogen Supply Pressure Low Alarm
10. AOP-5A Loss of Condenser Vacuum
11. PSA Section 7.3 Plant Flood Design Basis
12. PBNP Individual Plant Examination (IPE) for internal events and internal flood.
13. DG-C02, Internal Flooding
14. DBD-T41, Module A, "Hazards-Internal and External Flooding

HU2

Initiating Condition: FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.

Operating Mode Applicability: All

EAL:

HU2.1. FIRE in buildings or areas contiguous to any of the Table H-1 areas not extinguished within 15 minutes of control room notification or verification of a control room alarm

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

Basis:

The purpose of this IC is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, Detection is visual observation and report by plant personnel or sensor alarm indication. The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. Verification of a fire detection system alarm includes actions that can be taken with the control room or other nearby site-specific location to ensure that the alarm is not spurious. A verified alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket). This EAL excludes waste-basket FIREs and other small FIREs of no safety consequence.

Escalation to a higher emergency class is by IC HA4, "FIRE Affecting the Operability of Plant Safety Systems Required for the Current Operating Mode".

PBNP Basis Reference(s):

1. PBNP FSAR Table 3.3-1
2. Bechtel Drawing C-3 Plant Areas

HU3

Initiating Condition: Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.

Operating Mode Applicability: All

EAL:

HU3.1.	Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.
HU3.2.	Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

This IC is based on the existence of uncontrolled releases of toxic or flammable gas that may enter the site boundary and affect normal plant operations. It is intended that releases of toxic or flammable gases are of sufficient quantity, and the release point of such gases is such that normal plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. The EALs are intended to not require significant assessment or quantification. The IC assumes an uncontrolled process that has the potential to affect plant operations, or personnel safety.

Escalation of this EAL is via HA3, which involves a quantified release of toxic or flammable gas affecting VITAL AREAs.

PBNP Basis Reference(s):

1. PBNP FSAR Table 3.3-1

HU4

Initiating Condition: Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant.

Operating Mode Applicability: All

EAL:

HU4.1. Security Shift Supervisor reports ANY of the following:
- Suspected sabotage device discovered within the plant Protected Area
- Suspected sabotage device discovered outside the Protected Area or in the plant switchyard
- Confirmed tampering with safety-related equipment
- A hostage situation that disrupts NORMAL PLANT OPERATIONS
- Civil disturbance or strike which disrupts NORMAL PLANT OPERATIONS
- Internal disturbance that is not a short lived or that is not a harmless outburst involving ANY individuals within the Protected Area
- Malevolent use of a vehicle outside the Protected Area which disrupts NORMAL PLANT OPERATIONS

HU4.2 Notification of a credible site-specific threat by the Security Shift Supervisor or outside agency (e.g., NRC, military or law enforcement)

Basis:

Reference is made to the Security Shift Supervisor because this individual is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan.

HU4.1 is based on Physical Security Plan. Security events which do not represent a potential degradation in the level of safety of the plant, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Examples of security events that indicate Potential Degradation in the Level of Safety of the Plant are provided for consideration.

INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE would result in EAL escalation to an ALERT.

HU4.2 is to ensure that appropriate notifications for the security threat are made in a timely manner. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the Physical Security Plan.

A higher initial classification could be made based upon the nature and timing of the threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification in accordance with the Physical Security Plan and Emergency Plans.

PBNP Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. PBNP Physical Security Plan
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. NMC fleet Security Threat Assessment Policy

HU5

Initiating Condition: Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE.

Operating Mode Applicability: All

EAL:

HU5.1.	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Basis:

This EAL is intended to address 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 UE emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency response procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

PBNP Basis Reference(s):

None

HA1

Initiating Condition: Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability: All

EAL:

HA1.1. Seismic event GREATER THAN Operating Basis Earthquake (OBE) as indicated by:
Two or more seismic monitors indicate ground acceleration **EITHER:**
GREATER THAN OR EQUAL TO 0.06 g horizontal
OR
GREATER THAN OR EQUAL TO 0.04 g vertical

HA1.2. Tornado or high winds (15 minute average) GREATER THAN 108 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures / equipment or Control Room indication of degraded performance of those systems.

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

HA1.3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or control room indication of degraded performance of those systems:

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

HA1.4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas:

- Reactor Building
- Intake Building
- Refueling Water Storage Tank
- Diesel Generator Building
- Turbine Building
- Condensate Storage Tank
- Control Room

HA1.5. Uncontrolled flooding in following areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

Auxiliary building caused by rupture of the SW header

OR

Water intake structure caused by rupture of a circulating water system expansion joint or fire water main.

HA1.6 Lake (forebay) level GREATER THAN OR EQUAL TO 9.0 ft (588.2 ft IGLD)

Basis:

The EALs in this IC escalate from the UE EALs in HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other ICs (e.g., System Malfunction).

HA1.1 is based on the FSAR operating basis earthquake (OBE) of 0.06 g horizontal or 0.04 g vertical acceleration. Seismic events of this magnitude can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. [Ref. 1, 2, 3, 4]

HA1.2 is based on the FSAR design basis sustained wind speed of 108 mph. Wind loads of this magnitude can cause damage to safety functions. Wind speed is measured as the 15 minute average wind speed. This EAL addresses events that may have resulted in a plant VITAL AREA being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant structures, systems or equipment. [Ref. 5, 6, 7] The list of plant structures/equipment containing functions required for safe shutdown of the plant. [Ref. 2] Ultimate Heat Sink is not included in this list as there were no events identified which would adversely impact Lake Level [Ref. 13]

HA1.3 addresses crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. [Ref. 7] The list of plant structures/equipment containing functions required for safe shutdown of the plant. [Ref. 2] Ultimate Heat Sink is not included in this list as there were no events identified which would adversely impact Lake Level [Ref. 13]

HA1.4 addresses the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant. [Ref. 8, 9, 10] The list of plant structures/equipment containing functions required for safe shutdown of the plant. [Ref. 2] Ultimate Heat Sink is not included in this list as there were no events identified which would adversely impact Lake Level [Ref. 13]

HA1.5 addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. The auxiliary building and water intake structure are those areas identified in the IPE that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged. [Ref. 11, 12]

HA1.6 covers high lake (forebay) 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 GREATER THAN OR EQUAL TO 9.0 feet (588.2 ft elevation IGLD) corresponds to that which can result in flooding of vital areas containing safe shutdown equipment.

PBNP Basis Reference(s):

1. AOP-28 Seismic Event
2. FSAR Volume 1 Section 2.9 Seismology
3. STPT 22.1 Seismic Event Monitoring
4. EPRI document, "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989
5. FSAR Volume 3 Page 5.1-37 Wind and Tornado Forces
6. PSA Section 9 Notebook 9.1
7. Bechtel Drawing C-3 Plant Areas
8. ARP 1(2) C33 1-2 Hydrogen Pressure High-Low Alarm
9. ARP 1(2) C33 1-3 Hydrogen Supply Pressure Low Alarm
10. AOP-5A Loss of Condenser Vacuum
11. PSA Section 7.3 Plant Flood Design Basis
12. PBNP Individual Plant Examination (IPE) for internal events and internal flood.
13. FPER, Fire Protection Evaluation Report

HA2

Initiating Condition: FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

EAL:

HA2.1. FIRE or EXPLOSION in any of the following areas (Table H-1):

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

AND

Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.

Basis:

These areas contain systems and components required for the safe shutdown functions of the plant. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs.

This EAL addresses a FIRE / EXPLOSION and not the degradation in performance of affected systems. System degradation is addressed in the System Malfunction EALs. The reference to damage of systems is used to identify the magnitude of the FIRE / EXPLOSION and to discriminate against minor FIREs / EXPLOSIONs. The reference to safety systems is included to discriminate against FIREs / EXPLOSIONs in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE / EXPLOSION was large enough to cause damage to these systems.

The inclusion of a "report of VISIBLE DAMAGE" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The Emergency Director also needs to consider any security aspects of the EXPLOSIONs, if applicable.

PBNP Basis Reference(s):

1. PSA Section 8.0 Fire Hazards Analysis Table 8.1.3-1 Safe Shutdown Systems

HA3

Initiating Condition: Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

EAL:

HA3.1. Report or detection of toxic gases within or contiguous to a VITAL AREA (Table H-1) in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

HA3.2. Report or detection of gases in concentration GREATER THAN the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA (Table H-1).

Table H-1 VITAL AREAS
• 1(2) Containment Building • Primary auxiliary building • Turbine Building • Control Building • Diesel Generator Building • Gas Turbine Building • Circ Water Pump House

Basis:

This IC is based on gases that affect the safe operation of the plant. This IC applies to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radioactive Effluent, or Emergency Director Judgment ICs.

HA3.1 is met if measurement of toxic gas concentration results in an atmosphere that is IDLH within a VITAL AREA or any area or building contiguous to VITAL AREA. Exposure to an IDLH atmosphere will result in immediate harm to unprotected personnel, and would preclude access to any such affected areas.

HA3.2 is met when the flammable gas concentration exceeds the LOWER FLAMMABILITY LIMIT. This EAL addresses concentrations at which gases can ignite/support combustion. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury.

PBNP Basis Reference(s):

1. PSA Section 8.0 Fire Hazards Analysis Table 8.1.3-1 Safe Shutdown Systems

HA4

Initiating Condition: Confirmed Security Event in a Plant PROTECTED AREA.

Operating Mode Applicability: All

EAL:

HA4.1. INTRUSION into the PROTECTED AREA by a HOSTILE FORCE.

HA4.2. Security Shift Supervisor reports any of the following:

- Sabotage device discovered in the PROTECTED AREA
- Standoff attack on the site protected area by a HOSTILE FORCE (i.e., Sniper)
- ANY Security event of increasing severity that persists for GREATER THAN 30 minutes:
  - o Credible bomb threats
  - o Extortion
  - o Suspicious Fire or Explosion
  - o Significant Security System Hardware Failure
  - o Loss of Guard Post Contact

Basis:

This class of security events represents an escalated threat to plant safety. A confirmed INTRUSION report is satisfied if physical evidence indicates the presence of a HOSTILE FORCE within the PROTECTED AREA.

The Physical Security Plan identifies numerous events/conditions that constitute a threat/compromise to station security.

INTRUSION into a VITAL AREA by a HOSTILE FORCE will escalate this event to a Site Area Emergency.

Reference is made to Security Shift Supervisor because this individual is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy, SE 0018

HA5

Initiating Condition: Control Room Evacuation Has Been Initiated.

Operating Mode Applicability: All

EAL:

HA5.1. Entry into AOP-10 Control Room Inaccessibility for control room evacuation.

Basis:

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facility is necessary. The AOP-10 series of procedures provide specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations. Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

PBNP Basis Reference(s):

1. AOP-10 Control Room Inaccessibility
2. AOP-10A Safe Shutdown - Local Control
3. AOP-10B Safe to Cold Shutdown in Local Control

HA6

Initiating Condition: Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.

Operating Mode Applicability: All

EAL:

HA6.1.	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

This EAL is intended to address 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 Alert emergency class.

PBNP Basis Reference(s):

1. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents

HS1

Initiating Condition: Confirmed Security Event in a Plant VITAL AREA.

Operating Mode Applicability: All

EAL:

HS1.1.	INTRUSION into the plant VITAL AREA by a HOSTILE FORCE.
HS1.2.	Security Supervision reports confirmed sabotage discovered in a VITAL AREA

Basis:

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC in that a HOSTILE FORCE has progressed from the PROTECTED AREA to a VITAL AREA.

Consideration is given to the following types of events when evaluating an event against the criteria of the site specific Physical Security Plan: SABOTAGE and HOSTAGE / EXTORTION. The Physical Security Plan identifies numerous events/conditions that constitute a threat/compromise to a Station's security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered.

Loss of Plant Control would escalate this event to a GENERAL EMERGENCY.

Reference is made to Security Shift Supervisor because this individual is the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan.

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy, SE 0018

HS2

Initiating Condition Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.

Operating Mode Applicability: All

EAL:

HS2.1.	Control room evacuation has been initiated. **AND** Control of the plant cannot be established per AOP-10A Safe Shutdown - Local Control within 15 minutes.

Basis:

Expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the time for transfer that the operator has control of the plant from the remote shutdown panel.

The AOP-10 series of procedures provide specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. These safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

Escalation of this event would be by Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Emergency Director Judgment ICs.

PBNP Basis Reference(s):

1. AOP-10 Control Room Inaccessibility
2. AOP-10A Safe Shutdown - Local Control
3. AOP-10B Safe to Cold Shutdown in Local Control
4. BG AOP-10A, Background Document

HS3

Initiating Condition Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.

Operating Mode Applicability: All

EAL:

HS3.1.	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

This EAL is intended to address 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 class description for Site Area Emergency.

PBNP Basis Reference(s):

1. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents

HG1

Initiating Condition: Security Event Resulting in Loss Of Physical Control of the Facility.

Operating Mode Applicability: All

EAL:

HG1.1. A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of

EITHER:

A VITAL AREA such that operation of equipment required for safe shutdown is lost

OR

Spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in the pool).

Basis:

This IC encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAs (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location. These safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink). If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the above initiating condition is not met.

This EAL addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in pool).

Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions.

PBNP Basis Reference(s):

1. Physical Security Plan
2. NMC fleet Security Threat Assessment Policy, SE 0018

HG2

Initiating Condition: Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.

Operating Mode Applicability: All

EAL:

HG2.1.	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL is intended to address 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 General Emergency class.

PBNP Basis Reference(s):

1. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents

SU1

Initiating Condition Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SU1.1. Loss of all offsite power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for GREATER THAN 15 minutes.

AND

At least 1 emergency generator is supplying power to an emergency bus.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

For the purpose of classification under this EAL, offsite power sources include any of the following:

- 345 KVAC system supplying power to the 13.8 KVAC system and the unit LVSATs
- Cross-tying with the opposite unit power supply
- Backfeeding power to the UATs through the 19 KVAC system and the main step-up transformer X-01.

The capability to cross-tie AC power from the other unit takes credit for the redundant power source for this IC. The inability to effect the cross-tie within 15 minutes warrants declaring a UE.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A Main Power Transformer Backfeed

SU2

Initiating Condition: Inability to Reach Required Shutdown Within Technical Specification Limits.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SU2.1.	Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the PBNP Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate UE is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of a UE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. PBNP Technical Specifications

SU3

Initiating Condition: UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SU3.1. UNPLANNED loss of most or all annunciators or indicators associated with the following safety systems for GREATER THAN 15 minutes

- ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

Basis:

This IC and EAL recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This is addressed by the specific Technical Specification.

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

Escalated to an Alert is appropriate if a transient is in progress during the loss of annunciation or indication.

PBNP Basis Reference(s):

1. OM 1.1, Conduct of Plant Operations, Attachment 2

SU4

Initiating Condition: Fuel Clad Degradation.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SU4.1.	Failed Fuel Monitor (RE-109) GREATER THAN 750 mR/hr indicating fuel clad degradation
SU4.2.	Coolant sample activity GREATER THAN 50 µCi/gm dose equivalent I-131 indicating fuel clad degradation.

Basis:

This IC and EALs is a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

SU4.1 addresses the failed fuel monitor, that provides indication of fuel clad integrity [Ref. 2]. 750 mR/hr is the value corresponding to approximately 0.1% fuel clad damage.

SU4.2 addresses coolant samples exceeding coolant technical specifications for iodine spike [Ref. 1].

Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs. The referenced Technical Specification limits are mode dependent.

PBNP Basis Reference(s):

1. Tech Spec 3.4.16 – RCS Specific Activity
2. Calc 2004-0019, Failed Fuel Monitor (RE-109) Reading/Fuel Damage Correlation

SU5

Initiating Condition: RCS Leakage.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SU5.1.	Unidentified or pressure boundary leakage GREATER THAN 10 gpm.
SU5.2.	Identified leakage GREATER THAN 25 gpm.

Basis:

This IC may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

The 10 gpm value for SU5.1 was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

The 25 gpm value for SU5.2 is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.

Escalation to the Alert level is via Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. TS 3.4.13, RCS Operational Leakage limits
2. OI 55, Primary Leak Rate Calculation
3. OM 3.19, Reactor Coolant System Leakage Determination

SU6

Initiating Condition: UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability:
1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Plant Public Address System
• Security Radio
• Commercial Phone System
• Portable radios
• Sound power phones

SU6.2. Loss of all Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• Emergency Notification System (ENS)
• Health Physics Network (HPN)
• Operations Control Counterpart Link (OCCL)
• Management Counterpart Link (MCL)
• Protective Measures Counterpart Link (PMCL)
• Reactor Safety Counterpart Link (RSCL)
• Nuclear Accident Reporting System (NARS)
• Commercial Phone System
• General Telephone Lines
• Manitowoc County Sheriff's Department Radio

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems.

Table C-1 onsite communications loss [Ref. 1, 2] encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system (Gaitronics) and radios / walkie talkies).

Table C-2 offsite communications loss [Ref. 1, 2] encompasses the loss of all means of communications with offsite authorities.

PBNP Basis Reference(s):

1. EPMP 2.1, Testing of Communications Equipment
2. EPMP 2.1A, Monthly Communications Test

SU8

Initiating Condition: Inadvertent Criticality.

Operating Mode Applicability: 3-Hot Standby
4-Hot Shutdown

EAL:

SU8.1.	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses inadvertent criticality events. While the primary concern of this IC is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling IC is CU8.

The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements such as shutdown bank withdrawal.

Escalation would be by the Fission Product Barrier Matrix, as appropriate to the operating mode at the time of the event, or by Emergency Director Judgment.

Note: This EAL is SU8 following SU6. SU7 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

PBNP Basis Reference(s):

1. OP 1B, Reactor Startup

SA2

Initiating Condition: Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was Successful.

Operating Mode Applicability:
1-Power Operation
2-Startup
3-Hot Standby

EAL:

SA2.1. Indication(s) exist that a Reactor Protection System (RPS) setpoint was exceeded

AND

RPS automatic trip did NOT reduce power to LESS THAN 5%

AND

Any of the following operator actions are successful in reducing power to LESS THAN 5%:

- Use of Reactor Trip Buttons
- De-energizing 1(2)B01 and 1(2)B02

Basis:

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS integrity.

Failure of the automatic protection system is the issue. A manual scram is any set of actions by the reactor operator(s) at the reactor control console which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip button). Failure of manual scram would escalate the event to a Site Area Emergency.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees
2. BG-CSP-ST.0, Critical Safety Function Status Trees
3. EOP-0, Reactor Trip or Safety Injection

SA4

Initiating Condition: UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SA4.1. UNPLANNED loss of most or all annunciators or indicators associated with the following safety systems for GREATER THAN 15 minutes

- ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

AND

Either of the following: (a or b)

a. A SIGNIFICANT TRANSIENT is in progress.

OR

b. Compensatory non-alarming indications are unavailable.

Basis:

This IC and its associated EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

SIGNIFICANT TRANSIENT includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Compensatory non-alarming indications include the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

Escalation to a Site Area Emergency is warranted when the operating crew cannot monitor the transient in progress.

Note: This EAL is SA4 following SA2. SA2 is not used in NEI 99-01 Revision 4 and that convention is carried forward here.

PBNP Basis Reference(s):

1. OM 1.1, Conduct of Plant Operations, Attachment 2

SA5

Initiating Condition: AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SA5.1. AC power capability to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 reduced to only one of the following sources for GREATER THAN 15 minutes:

- A single emergency diesel generator (G01, G02, G03 or G04)
- LVSAT 1(2)-X04
- Cross-tying with the opposite unit power supply

AND

Any additional single failure will result in station blackout.

Basis:

The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout.

The subsequent loss of the single remaining power source would escalate the event to a Site Area Emergency via SS1.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A Unit 1, Main Power Transformer Backfeed

SS1

Initiating Condition: Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SS1.1. Loss of all **offsite** power to safety-related 4160 VAC buses 1(2)-A05 AND 1(2)-A06

AND

Failure of all emergency generators to supply power to safety-related 4160 VAC buses.

AND

Failure to restore power to at least one safety-related 4160 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency.

Escalation to General Emergency is via Fission Product Barrier Degradation or SG1.

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.

PBNP Basis Reference(s):

1. DBD-22, 4160 VAC System
2. DBD-18, 13.8 KVAC System
3. ECA 0.0, Loss of All AC Power
4. FSAR Section 8, Electrical Systems
5. AOP-14A U1, Main Power Transformer Backfeed

SS2

Initiating Condition: Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was NOT Successful.

Operating Mode Applicability: 1-Power Operation
2-Startup

EAL:

SS2.1.	Indication(s) exist that automatic <u>and</u> manual trip were NOT successful in reducing power to LESS THAN 5%.

Basis:

Automatic and manual trip are not considered successful if action away from the reactor control console was required to trip the reactor.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS.

Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Emergency Director Judgment ICs.

Automatic or manual reactor trip is considered successful if actions taken at the main control panels (use of reactor trip buttons, de-energizing 1(2)B01 or 1(2)B02) result in reducing reactor power less than 5%.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees
2. BG-CSP-ST.0, CRITICAL SAFETY FUNCTION STATUS TREES
3. CSP-S.1, Response to Nuclear Power Generation/ATWS
4. EOP-0, Reactor Trip or Safety Injection

SS3

Initiating Condition: Loss of All Vital DC Power.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SS3.1.	Loss of all vital DC power based on LESS THAN 115 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for GREATER THAN 15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

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

At 115 VDC, loss of equipment function would begin to occur. By the time the bus voltage has degraded to below 105 VDC, the ability to monitor and control important plant safety functions would be severely compromised. 115 VDC represents a value at which some plant functions would begin to be lost.

Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation, or Emergency Director Judgment ICs.

PBNP Basis Reference(s):

1. 0-SOP-DC-001/2/3/4

SS4

Initiating Condition: Complete Loss of Heat Removal Capability.

Operating Mode Applicability:

1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SS4.1.	Loss of core cooling (CSP-C.1) **AND** heat sink (CSP-H.1).

Basis:

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

Escalation to General Emergency would be via Abnormal Rad Levels / Radiological Effluent, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

PBNP Basis Reference(s):

1. CSP-C.1, Response to Inadequate Core Cooling
2. CSP-H.1, Response to Loss of Secondary Heat Sink

SS6

Initiating Condition: Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.

Operating Mode Applicability:
1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SS6.1. Loss of most or all annunciators associated with the following safety systems

- ECCS
- Containment Isolation
- Reactor Trip
- Process or Effluent Radiation Monitors
- Electrical Distribution/Diesel Generators

AND

SIGNIFICANT TRANSIENT in progress.

AND

Compensatory non-alarming indications are unavailable.

AND

Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, and maintain containment intact are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the inability of the control room staff to monitor the plant response to a transient. A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public.

SIGNIFICANT TRANSIENT includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Compensatory non-alarming indications include the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room.

Indications needed to monitor safety functions necessary for protection of the public include control room indications, computer generated indications and dedicated annunciation capability. The specific indications are those used to monitor the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

Quantification of the number of annunciators and indicators is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected.

PBNP Basis Reference(s):

1. OM 1.1, Conduct of Plant Operations, Attachment 2

SG1

Initiating Condition: Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:
1-Power Operation
2-Startup
3-Hot Standby
4-Hot Shutdown

EAL:

SG1.1. Loss of all offsite power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06

AND

Failure of all emergency diesel generators to supply power to safety-related 4160 VAC buses.

AND

Either of the following: (a or b)

a. Restoration of at least one safety-related 4160 VAC bus within 4 hours is NOT likely

OR

b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by conditions requiring entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment.

The 4 hours to restore AC power is based on the site blackout coping analysis. Appropriate allowance for offsite emergency response including evacuation of surrounding areas should be considered.

This IC is specified to assure that timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly the need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent? (Refer to Table F-1 for more information.)
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

PBNP Basis Reference(s):

1. DBD-T-46, Section 3.1, Station Blackout
2. 10 CFR 50.63 and Regulatory Guide 1.155, Station Blackout
3. DBD-22, 4160 VAC System
4. DBD-18, 13.8 KVAC System
5. ECA 0.0, Loss of All AC Power
6. FSAR Section 8, Electrical Systems
7. AOP-14A U1, Main Power Transformer Backfeed
8. CSP-C.1, Response to Inadequate Core Cooling
9. CSP-C.2, Response to Degraded Core Cooling
10. FSAR Appendix A, "Station Blackout"

SG2

Initiating Condition: Failure of the Reactor Protection System to Complete an Automatic Trip and Manual Trip was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability: 1-Power Operation
2-Startup

EAL:

SG2.1. Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 5%.

AND

Either of the following: (a or b)

a. Indication(s) exists that the core cooling is extremely challenged Core Cooling - RED path (CSP-C.1).

OR

b. Indication(s) exists that heat removal is extremely challenged Heat Sink - RED path (CSP-H.1).

Basis:

Automatic and manual trip are not considered successful if action away from the reactor control console is required to trip the reactor.

Under the conditions of this IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence.

The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F or that the reactor vessel water level is below the top of active fuel. This EAL equates to a core cooling RED condition and an entry into a critical safety procedure (CSP-C.1).

The inability to initially remove heat during the early stages of this sequence is addressed. If emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist. This EAL equates to a Heat Sink RED condition and an entry into a critical safety procedure (CSP-H.1).

In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design (typically 3 to 5% power) a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

PBNP Basis Reference(s):

1. CSP-ST.0, Critical Safety Function Status Trees, Figures 1, 2 and 3
2. CSP-S.1, Response to Nuclear Power Generation/ATWS
3. CSP-C.1, Response to Inadequate Core Cooling
4. CSP-H.1, Response to Loss of Secondary Heat Sink

Table F-0

Recognition Category F

Fission Product Barrier Degradation

INITIATING CONDITION MATRIX

UE		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
FU1	ANY Loss or ANY Potential Loss of Containment *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	FA1	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	FS1	Loss or Potential Loss of ANY Two Barriers *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	FG1	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*

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. UE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

   - At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

   - The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily lowering would represent an increasing risk to public health and safety.

2. Fission Product Barrier ICs must be capable of addressing event dynamics. Thus, the EAL Reference Table F-1 states that imminent (i.e., within 2 hours) Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

TABLE F-1
PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
Conditions requiring entry into Core-Cooling RED Path (CSP-C.1)	Conditions requiring entry into Core Cooling-ORANGE Path (CSP-C.2) **OR** Conditions requiring entry into Heat Sink-RED Path (CSP-H.1)	Not Applicable	Conditions requiring entry into RCS Integrity-RED Path (CSP-P.1) **OR** Conditions requiring entry into Heat Sink-RED Path (CSP-H.1)	Not Applicable	Conditions requiring entry into Containment-RED Path (CSP-Z.1)
2. Primary Coolant Activity Level		**2. RCS Leak Rate**		**2. Containment Pressure**	
Coolant Activity GREATER THAN 300 µCi/gm I-131 equivalent	Not Applicable	GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling (LESS THAN OR EQUAL TO [80 degree F] 35 degree F)	Unisolable leak exceeding 60 gpm	Rapid unexplained lowering following initial rise **OR** Containment pressure or sump level response not consistent with LOCA conditions	60 PSIG and rising **OR** Hydrogen concentration in containment GREATER THAN OR EQUAL TO 6% **OR** pressure GREATER THAN 25 psig with LESS THAN one full train of depressurization equipment operating
3. Core Exit Thermocouple Readings				**3. Core Exit Thermocouple Reading**	
GREATER THAN OR EQUAL TO 1200 degree F	GREATER THAN OR EQUAL TO 700 degree F			Not applicable	Core exit thermocouples in excess of 1200 degrees F and restoration procedures not effective within 15 minutes; **OR,** core exit thermocouples in excess of 700 degrees F with reactor vessel level below 27 ft RVLIS NR (with no RCPs running) top of active fuel and restoration procedures not effective within 15 minutes

TABLE F-1
PBNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY Potential Loss of Containment	ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS	Loss or Potential Loss of ANY two Barriers	Loss of ANY two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		RCS Barrier EALS		Containment Barrier EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
4. Reactor Vessel Water Level		**3. SG Tube Rupture**		**4. SG Secondary Side Release with P-to-S Leakage**	
Not Applicable	Level LESS THAN OR EQUAL TO • 25 ft RVLIS NR (with no RCPs running) • [100 ft] 90 ft RVLIS WR (with 1 RCP running) • [120 ft] 110 ft RVLIS WR (with 2 RCPs running)	SGTR that results in an ECCS (SI) Actuation	Not Applicable	RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate greater than 10 gpm with nonisolable steam release from affected S/G to the environment	Not applicable
				5. CNMT Isolation Valves Status After CNMT Isolation	
				Containment isolation valve(s) not closed AND Downstream pathway to the environment exists	Not Applicable
5. Containment Radiation Monitoring		**4. Containment Radiation Monitoring**		**6. Significant Radioactive Inventory in Containment**	
Containment rad monitor reading GREATER THAN 17 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128	Not Applicable	Containment rad monitor reading GREATER THAN 3.5 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128	Not Applicable	Not Applicable	Containment rad monitor reading GREATER THAN 15,900 R/hr indicated on any of the following: • 1(2) RM-126 • 1(2) RM-127 • 1(2) RM-128
6. Other Indications		**5. Other Indications**		**7. Other Indications**	
Failed Fuel Monitor (RE-109) reading GREATER THAN OR EQUAL TO 4500 mrem/hr	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable
7. Emergency Director Judgment		**6. Emergency Director Judgment**		**8. Emergency Director Judgment**	
Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barier		Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier		Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	

Basis Information For Table F-1
PBNP Emergency Action Level
Fission Product Barrier Reference Table

FUEL CLAD BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6 or 7)

The Fuel Clad Barrier is the zircalloy or stainless steel tubes that contain the fuel pellets.

1. Critical Safety Function Status

RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.
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. [Ref. 1, 2]

Heat Sink - RED indicates the ultimate heat sink function is under extreme challenge and thus these two items (Core Cooling – ORANGE or Heat Sink – RED) indicate potential loss of the Fuel Clad Barrier. CSP-H.1 is the Critical Safety Procedure that provides directions to respond to a loss of secondary heat sink in both steam generators. [Ref. 1, 3]

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

CSFST setpoints enclosed in brackets (e.g., [120 ft], etc.) are used under adverse containment conditions. Adverse containment conditions are defined as:

- Containment pressure is equal to or greater than 10 psig.
- Containment radiation is currently greater than or equal to 1E5 R/hr.
- Integrated dose is greater than 1E6 R or unknown.

CSP-H.1 is the Critical Safety Procedure that provides directions to restore core cooling. [Ref. 1, 1]

The barrier loss/potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

2. Primary Coolant Activity Level

This value is 300 μCi/gm I_{131} equivalent. Assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no equivalent "Potential Loss" EAL for this item.

3. Core Exit Thermocouple Readings

Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked).

The "Loss" EAL 1200 degrees F reading should correspond to significant superheating of the coolant. This value corresponds to the temperature reading that indicates core cooling - RED in Fuel Clad Barrier EAL #1 which is 1200 degrees F. [Ref. 1]

The "Potential Loss" EAL 700 degrees F reading should correspond to loss of subcooling. This value corresponds to the temperature reading that indicates core cooling - ORANGE in Fuel Clad Barrier EAL #1 which is 700 degrees F. [Ref.1]

4. Reactor Vessel Water Level

There is no "Loss" EAL corresponding to this item because it is better covered by the other Fuel Clad Barrier "Loss" EALs.

The "Potential Loss" EAL is defined by the Core Cooling - ORANGE path [Ref.1]. This water level is a consideration in the Core Cooling - ORANGE path only after the determination is made that no RCPs are running.

5. Containment Radiation Monitoring

The 17 R/hr reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere. [Ref. 6, 7] Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier Loss EAL #4. Thus, this EAL indicates a loss of both the fuel clad barrier and a loss of RCS barrier.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

Potential Loss - Not applicable

6. Failed Fuel Monitor (RE-109) reading GREATER THAN 4500 mRem/hr

A Failed Fuel Monitor reading of greater than 4500 mR/hr indicates the release of reactor coolant, with elevated activity indicative of fuel damage. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 activity into the Primary System. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage. [Ref. 8]

There is no "Potential Loss" EAL associated with this item.

7. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

RCS BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6)

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

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

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. [Ref. 9, 10]

Heat Sink-Red path is entered if narrow range level in any S/G is equal to or less than [51%] 29% and total feedwater flow to S/Gs is equal to or less than 200 gpm. The combination of these two conditions indicates 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. [Ref. 11]

The barrier potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

Loss - Not applicable

2. RCS Leak Rate

The "Loss" EAL addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one charging pump discharging to the charging header. A second charging pump being required is indicative of a substantial RCS leak. 60 gpm is the minimum operability flow rate for each charging pump. [Ref. 12]

3. SG Tube Rupture

This EAL is intended to address the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier "Loss" EAL #4 and Fuel Clad Barrier EALs. The "Loss" EAL

addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation of ECCS (SI). ECCS (SI) actuation is caused by:

- PZR Low Pressure (equal to or less than 1735 psig)
- Steam Line Low Pressure (equal to or less than 530 psig)
- Containment High Pressure (equal to or greater than 5 psig)

140 gpm is the design maximum capacity of all charging pumps.

This is consistent to the RCS Barrier "Potential Loss" EAL #2. This condition is described by "entry into E-3 required by EOPs". By itself, this EAL will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment Barrier "Loss" EAL #4. [Ref. 14, 15]

There is no "Potential Loss" EAL.

4. Containment Radiation Monitoring

The 3.5 R/hr reading is a value which indicates the release of reactor coolant to the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the containment atmosphere. [Ref. 6, 7] This reading is less than that specified for Fuel Clad Barrier EAL #5. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL #5, fuel damage would also be indicated.

Monitors used for this fission product barrier loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

Potential Loss - Not applicable

5. Other (Site-Specific) Indications

None

6. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

CONTAINMENT BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6 or 7 or 8)

The Containment Barrier includes the containment building, its 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.

1. Critical Safety Function Status

RED path indicates an extreme challenge to the safety function. Containment-Red path is entered if containment pressure is equal to or greater than 60 psig. This pressure is the containment design pressure and is well in excess of that expected from the design basis loss of coolant accident, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier. CSP-Z.1 is the Critical Safety Procedure that provides directions to respond to high containment pressure. [Ref. 16, 17, 18]

The barrier potential loss occurs when the plant parameter associated with the CSFST path is reached (not when the operator reads the CSFST in the EOP network). The phrase "Conditions requiring entry into..." is included in these thresholds to emphasize this intent.

Loss - Not applicable

2. Containment Pressure

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure rise indicates a loss of containment integrity. FSAR Figure 14.3.2-1 illustrates containment pressure response for a bounding LOCA. Containment pressure peaks at approximately 34 psia. [Ref. 25, 26, 27, 28]

Containment pressure and sump levels should rise as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure not rising indicates containment bypass and a loss of containment integrity.

The 60 PSIG for potential loss of containment is based on the containment design pressure. [Ref. 24, 25, 26, 27]

If hydrogen concentration reaches or exceeds 6% in an oxygen rich environment, an explosive mixture exists. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred As described above, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier. [Ref. 21, 22]

The second potential loss EAL represents a potential loss of containment in that the containment heat removal/depressurization system (but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint (25 psig) at which the equipment was supposed to have actuated. 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. [Ref. 16, 20, 23]

3. Core Exit Thermocouples

In this EAL, the restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is lowering or if the vessel water level is rising.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

The conditions in this potential loss EAL represent an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the Core Cooling and Heat Sink criteria in the Fuel and RCS barrier columns, this EAL would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path. [Ref. 1, 5]

Loss - Not applicable

4. SG Secondary Side Release With Primary To Secondary Leakage

This "loss" EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The first "loss" EAL addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers. In conjunction with RCS Barrier "loss" EAL #3, this would always result in the declaration of a Site Area Emergency. A faulted S/G means the existence of secondary side leakage that results in an uncontrolled lowering in steam generator pressure or the steam generator being completely depressurized. A ruptured S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection. Confirmation should be based on diagnostic activities consistent with EOP-0, Reactor Trip or Safety Injection. [Ref. 14]

The second "loss" EAL addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs. [Ref. 14]

It should be realized that the two "loss" EALs described above could be considered redundant. This was recognized during the development process. The inclusion of an EAL that uses Emergency Procedure

commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

A pressure boundary leakage of 10 gpm is used as the threshold in IC SU5.1, RCS Leakage, and is deemed appropriate for this EAL. For smaller breaks, not exceeding the normal charging capacity threshold in RCS Barrier "Potential Loss" EAL #2 (RCS Leak Rate) or not resulting in ECCS actuation in EAL #3 (SG Tube Rupture), this EAL results in a UE. For larger breaks, RCS barrier EALs #2 and #3 would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this EAL would exist in conjunction with RCS barrier "Loss" EAL #3 and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

5. Containment Isolation Valve Status After Containment Isolation

This EAL is intended to address incomplete containment isolation that allows direct release to the environment. It represents a loss of the containment barrier.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

Potential Loss - Not applicable

6. Significant Radioactive Inventory in Containment

The 15,900 R/hr reading is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS Barriers. [Ref. 6, 7] A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

Monitors used for this fission product barrier potential loss threshold are the containment high-range area monitors:

- 1(2) RM-126
- 1(2) RM-127
- 1(2) RM-128

Loss - Not applicable

7. Other (Site-Specific) Indications

None

8. Emergency Director Judgment

Loss - Not applicable

7. Other (Site-Specific) Indications

None

8. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the CSFSTs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

PBNP Basis Reference(s):

1. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
2. CSP-C.2, Response to Degraded Core Cooling
3. CSP-H.1, Response to Loss of Secondary Heat Sink
4. Volian Enterprises Calculation No. WEP-SPT-25, Reactor Vessel Level EOP Setpoints
5. CSP-C.1, Response to Inadequate Core Cooling
6. PBF 1608, Calculation 2004-0006, Dose Rate Calculation for Containment High Range and Refueling Floor Area Radiation Monitors Under Accident Conditions
7. SAMG SAG-5, Reduce Fission Product Releases, Attachment D
8. CALC 2004-0019, Failed Fuel Monitor (RE-109) Reading/Fuel Damage Correlation
9. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 4
10. CSP-P.1, Response to Imminent Pressurized Thermal Shock Condition
11. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 3, Heat Sink
12. DBD-04, Chemical and Volume Control System, Section 3.9
13. BG-CSP-ST.0 Step ST-2, Critical Safety Status Trees
14. EOP-0, Reactor Trip Or Safety Injection
15. DBD-04, Chemical and Volume Control System
16. CSP-ST.0, Unit 1(2) Critical Safety Function Status Trees, Figure 5
17. BG-CSP-ST.0 Step ST-5
18. CSP-Z.1, Response to High Containment Pressure
19. FSAR Section 5.1, Containment System Structure
20. BG-CSP-ST.0, CSFST, Step F.0.5
21. CSP-C.1 Unit 1 Red, Critical Safety Procedure Safety Related Response To Inadequate Core Cooling, Step 11

28. CSP-Z.1, Attachment B, Containment Isolation Valves
29. CALC WEP-SPT-12, EOP S1 Reduction Analysis 6/25/99
30. STPT 2.1, Safety Injection, Rev 2., dated 10/15/96