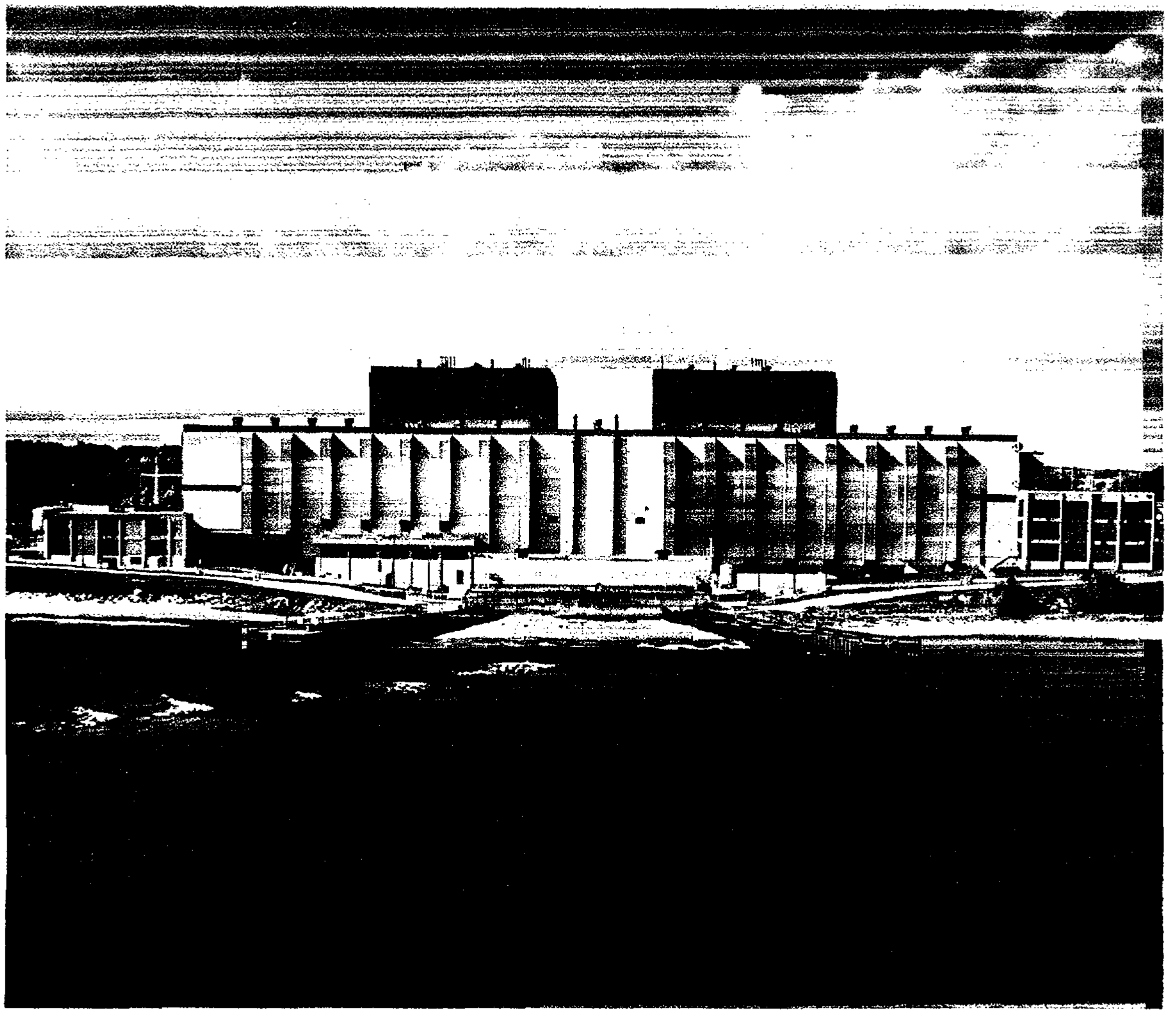
ENCLOSURE I



CURRENT POINT BEACH
EAL SCHEME

NUREG 0654

PEN/INK CHANGES

EMERGENCY ACTION LEVEL (EAL) OVERVIEW MATRIX

	CATEGORY	UNUSUAL EVENT	ALERT	SITE EMERGENCY	GENERAL EMERGENCY
Fission Product Barriers	See Also Attachment C	1.1.1.1 Reactor coolant sample activity greater than Technical Specification TS 3.4.16 1.1.2.1 Fuel monitor [1(2) RE-109] reading greater than 120 mRem/hr OR 2 of 3 containment high radiation monitors read greater than 1000 R/hr. 1.1.3.1 Primary to secondary leakage greater than Technical Specification Reference TS 3.4.13.d (500 gallons per day in either steam generator). 1.1.4.1 Unisolable primary system leakage greater than Technical Specification Reference TS 3.4.13.c (10 gallons per minute). 1.1.5.1 Excess RCS cooldown OR cold overpressurization of the RCS (ST-4 Integrity Orange path.)	1.1.1.2 Exceeding the LOSS threshold of either Fuel Clad OR Reactor Coolant System (RCS) barrier based on FPB Matrix (See Attachment C for thresholds). 1.1.2.2 Unisolable steam line break outside containment with greater than 10 gpm, but less than 50 gpm, primary to secondary leakage.	1.1.1.3 Exceeding the LOSS threshold of any 2 fission product barriers based on FPB Matrix (See Attachment C for thresholds).	1.1.1.4 Exceeding the LOSS threshold of any 2 fission product barriers AND exceeding the loss or challenge threshold of the 3rd barrier based on FPB Matrix (See Attachment C for thresholds).
System Malfunctions	2.1 Failure to Trip		2.1.1.2 Failure of the reactor protection system (automatic or manual) to initiate and complete a trip which brings the reactor subcritical.	2.1.1.3 Failure to rapidly bring the reactor subcritical from the Control Room (ST-1 Subcriticality Red Path).	
	2.2 Technical Specification Requirements	2.2.1.1 Failure to reach Technical Specification required operating mode or condition within the specified time limit of the LCO action statement.			
	2.3 Loss of Indications / Communications	2.3.1.1 Unplanned loss of most (approximately 75%) safety system annunciators or indications on Control Room Panels for greater than 15 minutes AND increased monitoring is required for safe plant operation. 2.3.2.1 Loss of all communications capability affecting the ability to either: Perform routine operations. OR Notify offsite agencies or personnel.	2.3.1.2 Unplanned loss of most (approximately 75%) safety system annunciators or indications on Control Room Panels for greater than 15 minutes AND increased monitoring is required for safe plant operation. AND either: A significant plant transient in progress. OR Plant Process Computer is unavailable.	2.3.1.3 Unplanned loss of most (approximately 75%) safety system annunciators or indications on Control Room Panels. AND Loss of ability to monitor critical safety function status. AND A significant plant transient in progress.	
	2.4 Safety System Performance		2.4.1.2 Inability to maintain reactor coolant temperature less than or equal to 200°F.	2.4.1.3 Primary to secondary leakage greater than 400 gpm. AND Inability to power BOTH buses A-05 AND A-06 from offsite sources.	
	2.5 Reactivity Transient	2.5.1.1 Uncontrolled Rod Withdrawal (FSAR 14.1.1 & 14.1.2)			
	2.6 Feedwater Transient				2.6.1.4 Transient initiated by loss of feedwater, followed by loss of auxiliary feedwater for > 1 hour. As indicated by ALL of the following: 1. Decreasing SG Levels- "A" SG [LI-461, LI-462, LI-463] "B" SG [LI-471, LI-472, LI-473] 2. No auxiliary feedwater flow- [FI-4002, FI-4007, FI-4014] [FI-4036, FI-4037]
Electrical Power	3.1 Loss of Vital AC Power	3.1.1.1 Loss of all offsite AC capability to vital buses as indicated by the inability to power BOTH buses A-05 AND A-06 of a given unit from offsite sources for greater than 15 minutes. 3.1.2.1 Loss of all on-site AC power capability to power BOTH buses A-05 AND A-06 of a given unit from onsite sources (GO1 through GO4) for greater than 15 minutes.	3.1.1.2 Loss of all safeguard bus AC power of a given unit as indicated by the inability to power BOTH buses A-05 AND A-06 OR B-03 AND B-04 AND Loss is for less than 15 minutes.	3.1.1.3 Loss of all safeguard bus AC power of a given unit as indicated by the inability to power BOTH buses A-05 AND A-06 OR B-03 AND B-04 AND Loss is for greater than 15 minutes.	3.1.1.4 Loss of all safeguard bus AC power of a given unit as indicated by the inability to power BOTH buses A-05 AND A-06 OR B-03 AND B-04 AND Loss is greater than 15 minutes AND Both narrow range S/G level less than [51%] 29% AND total feedwater flow to S/Gs less than 200 gpm (ST-3 Heat Sink Red Path.)
	3.2 Loss of Vital DC Power		3.2.1.2 Loss of all vital DC power as indicated by less than 105 vdc on all station battery buses (D01, D02, D03, D04) for less than 15 minutes.	3.2.1.3 Loss of all vital DC power as indicated by less than 105 vdc on all station battery buses (D01, D02, D03, D04) for greater than 15 minutes.	
Radiological	4.1 Off-site Radiological Release	4.1.1.1 Vent radiation reading(s) exceed the high alarm setpoints for greater than 60 minutes. OR Liquid release in excess of alarm setpoints which cannot be isolated.	4.1.1.2 Vent radiation readings exceed ten times the high alarm setpoints for greater than 15 minutes. OR Liquid release in excess of ten times alarm setpoints which cannot be isolated.	4.1.1.3 a. Effluent monitors detect levels corresponding to either: (1) 0.1 Rem Total Effective Dose Equivalent (TEDE). (2) 0.5 Rem thyroid Committed Dose Equivalent (CDE) at the site boundary under actual meteorological conditions. b. Either of the above doses measured in the environs. c. Either of the above doses projected based on plant parameters.	4.1.1.4 a. Effluent monitors detect levels corresponding to either: (1) 1 Rem Total Effective Dose Equivalent (TEDE). (2) 5 Rem thyroid Committed Dose Equivalent (CDE) at the site boundary under actual meteorological conditions. b. Either of the above doses measured in the environs. c. Either of the above doses projected based on other plant parameters
	4.2 In-Plant Radiological Conditions		4.2.1.2 Loss of control of radioactive material resulting in area radiation exceeding 1000X normal or expected levels within the Protected Area. Normal may be determined by trend recorder or other relevant data.		
Internal Events	5.1 Security Threats	5.1.1.1 Bomb, credible bomb threat, indication of sabotage, or attempted entry into the Protected Area by a hostile force.	5.1.1.2 Intrusion into the Protected Area by a hostile force.	5.1.1.3 Intrusion into a plant Vital Area by a hostile force.	5.1.1.4 A Security Event which results in either: Loss of physical control of the Control Room. OR Loss of remote shutdown capability.
	5.2 Control Room Habitability		5.2.1.2 Evacuation of the Control Room has been initiated with control of shutdown systems established from local stations.	5.2.1.3 Evacuation of the Control Room without establishment of plant control from remote shutdown stations within approximately 15 minutes.	
	5.3 Fire / Explosion	5.3.1.1 Near or on-site explosion as reported to Shift Manager by plant personnel making visual observation. 5.3.2.1 Fire within the Protected Area lasting more than 10 minutes after use of fire extinguishing equipment.	5.3.1.2 Explosion affecting operability of one (1) train of safety systems. 5.3.2.2 Fire affecting operability of one (1) train of a safety system.	5.3.1.3 Explosion affecting operability of two (2) trains of safety systems. 5.3.2.3 Fire affecting operability of two (2) trains of safety systems.	
	5.4 Turbine Rotating Component Failures	5.4.1.1 Visual confirmation of turbine housing penetration by a blade or rotating component.			
External Events	6.1 Natural Destructive Phenomena	6.1.1.1 Any earthquake felt by Control Room Operators. OR An indicator light on two or more of the following Seismic Event Monitors: SEI-6210 #3 Warehouse. SEI-6211 Unit 1 Façade. SEI-6212 Drum Prep Room. SEI-6213 El. 8' between vital switchgear room and Aux Feedwater tunnel. 6.1.2.1 Any tornado visible from the site.	6.1.1.2 Valid Seismic Event Monitor readings of an intensity greater than 0.04g vertical or 0.06g horizontal. 6.1.2.2 Indications or observations that a tornado has damaged a vital structure. OR Sustained winds greater than 90 MPH.	6.1.1.3 Valid Seismic Event Monitor readings of an intensity greater than 0.08g vertical or 0.12g horizontal. 6.1.2.3 Sustained winds greater than 100 MPH AND Reports or indications of damage to vital equipment or structures.	
	6.2 High Lake / Low Forebay Water Level	6.2.1.1 Less than -11' forebay or pump bay level with one unit's CW pumps off. 6.2.2.1 Any flooding which precludes access to the site or areas of the plant.	6.2.2.2 Flooding as indicated by greater than 6" of water in the 8' elevation of the Turbine Building.	6.2.2.3 Greater than 2' of water in vital switchgear room OR Greater than 2' of water in auxiliary feedwater pump room	
	6.3 Toxic / Flammable Gas Intrusion	6.3.1.1 Near or on-site flammable or toxic gas release as reported to Shift Manager by plant personnel making visual observation.	6.3.1.2 Entry of toxic or flammable gas into a plant building atmosphere affecting operation or access.	6.3.1.3 Entry of toxic or flammable gas into a plant vital area affecting operation or personnel safety AND Reactor coolant temperature greater than 200°F.	
	6.4 Vehicle / Missile Impacts	6.4.1.1 Unusual aircraft activity over facility.	6.4.1.2 Aircraft crash in protected area (within the fence). 6.4.2.2 Missile impact from any source by visual observation of Operations Supervisor.	6.4.1.3 Aircraft crash affecting operability of two (2) trains of safety systems. 6.4.2.3 Any missile impact affecting operability of two (2) trains of safety systems.	
Fuel Handling/ISFSI Events	7.1 Fuel Handling Events		7.1.1.2 Report of possible damage to irradiated fuel combined with an alarm on any of the following radiation monitors: RE-211, Containment air particulate monitor. RE-212 Containment noble gas monitor. RE-221 Drumming Area Vent. Manipulator Area Monitor. Spent Fuel Bridge Area Monitor.		
	7.2 Irradiated Fuel Events		7.2.1.2 Indications of irradiated fuel uncovered.		
	7.3 ISFSI Events	7.3.1.1 A loaded spent fuel cask dropped or tipped.	7.3.1.2 Breach of a loaded spent fuel cask as indicated by a reading greater than 1000 mr/hr at 1 meter.		
Judgment		8.1.1.1 Any event which in the judgement of the Shift Manager or the Emergency Director could lead to, or has led to, a potential degradation of the level of safety of the plant.	8.1.1.2 Any event which in the judgement of the Shift Manager or the Emergency Director could cause or has caused actual or potential substantial degradation of the level of safety of the plant.	8.1.1.3 Any event which in the judgement of the Shift Manager or the Emergency Director could indicate actual or likely major failures of plant functions needed to protect the public. Any releases are not expected to result in exposures in excess of EPA PAGs.	8.1.1.4 Any event which in the judgement of the Shift Manager or the Emergency Director could lead to actual or imminent core damage and the potential for a large release of radioactive material (in excess of EPA PAGs) outside the site boundary.

REPLACED BY NEJ 99-01 Rev 1

EPIP 1.2

EMERGENCY CLASSIFICATION

DOCUMENT TYPE:	Technical
CLASSIFICATION:	NNSR
REVISION:	42
EFFECTIVE DATE:	March 10, 2004
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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TABLE OF CONTENTS

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). ~~He~~ 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 ~~p~~ P rotective A ~~a~~ction R ~~r~~ecommendations (PARS).

3.2 The notification to the NRC shall be completed immediately following state and count~~ies~~ y notifications and should not exceed~~ing~~ 60-minutes from event classification, event termination, or change in ~~(protective action recommendations~~ P A R {PARS}

~~3.3 Category 8 EALs (Judgment) provide the ability to classify any set of plant conditions based on the Emergency Class definitions, based on NUREG-0654 Appendix 1.~~

3.~~4~~ 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.~~5~~4 Monitor ~~Continuously reference both~~ plant conditions and the EALs ~~in this procedure~~ for potential re-classification.

3.~~6~~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

~~TIME / INITIALS~~

~~**NOTE:** **A large version of Attachment A is available in the Control Room, TSC, and EOF.**~~

~~5.1.1 Determine the category (or categories) of the event. (Column 1 of Attachment A). The categories are:~~ ___ / ___

~~1. Fission Product Barriers~~ ☐
~~2. System Malfunction~~ ☐
~~3. Electrical Power~~ ☐
~~4. Radiological~~ ☐
~~5. Internal Events~~ ☐
~~6. External Events~~ ☐
~~7. Fuel Handling/ISFSI Events~~ ☐
~~8. Emergency Director Judgment~~ ☐

TIME / INITIALS

NOTE: If the EAL relates to ~~Category 1 (~~Fission Product Barriers~~)~~, EPIP 1.2.1 ~~Attachment C~~ provides additional information on the CHALLENGE and LOSS criteria.

5.1.~~1~~1 Identify the status of Fission Product Barriers from Attachment ~~C,~~ A as required.

	Intact	Challenge	Loss
Fuel Clad			
RCS			
Containment			

_____ / _____

5.1.~~3~~2 Make an initial EAL selection from Attachment A.

_____ / _____

NOTE: A challenge to, ~~Do not "anticipate" challenge~~ 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.~~4~~3 Reference the individual EAL page(s) in EPIP 1.2.1, ~~Attachment B~~ for the EAL(s) selected. Read all fields on the page to determine/confirm that the EAL applies.

_____ / _____

~~5.1.5 Also reference the individual EAL pages for the next higher and lower emergency class - in that category (if such EALs exist). This should further confirm the initial selection and specific EAL.~~

_____ / _____

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.~~6~~4 **IF** an event has been categorized ~~on Attachment A,~~ and the threshold of the EAL and related ~~surrounding~~ conditions are verified to have been met or exceeded ~~(Attachments B and C),~~ **THEN declare the emergency.**

_____ / _____

a. Record the time of declaration, the emergency classification, and the EAL

Classification __________ EAL __________

_____ / _____

TIME / INITIALS

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.

/

~~**NOTE: IF this procedure is being implemented from the EOF, THEN verify Control Room is assisting with Gai-tronics announcements / evacuation alarm.**~~

c. ~~IF this procedure was entered from EPIP 1.1, Course of Actions,~~
~~THEN~~ 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). ~~This definition does~~ 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 (per NUREG-0654, II.A.1.d, II.B.2).

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.

/

TIME / INITIALS

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 ~~the actual declaration of the emergency is not required;~~ 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.

_____ / _____

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.

7.0 BASES

B-1 Code of Federal Regulation, 10 CFR 50

~~B-2 NUREG-0654/FEMA-REP-1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Plants, Revision 1, published November, 1980.(note:1)~~

~~B-3 NUMARC NESP-007, Methodology for Development of Emergency Actions Levels, Revision 2, January 1992.~~

~~B-4 U.S. Regulatory Commission Position Paper, Branch Position on Acceptable Deviations to Appendix 1 to NUREG-0654/FEMA-REP-1, dated July 11, 1994.~~

~~Note 1: With deviations allowed by "Branch Position on Acceptable Deviations to Appendix 1 to NUREG -0654 /FEMA REP-1 dated July 11,1994.~~

B-2 NEI 99-01/NUMARC NESP-007, METHODOLOGY FOR DEVELOPMENT OF EMERGENCY ACTION LEVELS, REVISION 4.

ATTACHMENT A
EMERGENCY ACTION LEVEL (EAL) OVERVIEW MATRIX

REPLACED BY NEI 99-01 Rev 4 SCHEME

CATEGORY		UNUSUAL EVENT	ALERT	SITE EMERGENCY	GENERAL EMERGENCY
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	5.4 Turbine Rotating Component Failures	5.4.1.1 Visual confirmation of turbine housing penetration by a blade or rotating component.			
External Events	6.1 Natural Destructive Phenomena	6.1.1.1 Any earthquake felt by Control Room Operators. OR An indicator light on two or more of the following Seismic Event Monitors: SEI-6210 #3 Warehouse. SEI-6211 Unit 1 Façade. SEI-6212 Drum Prep Room. SEI-6213 El. 8' between vital switchgear room and Aux Feedwater tunnel. 6.1.2.1 Any tornado visible from the site.	6.1.1.2 Valid Seismic Event Monitor readings of an intensity greater than 0.04g vertical or 0.06g horizontal. 6.1.2.2 Indications or observations that a tornado has damaged a vital structure. OR Sustained winds greater than 90 MPH.	6.1.1.3 Valid Seismic Event Monitor readings of an intensity greater than 0.08g vertical or 0.12g horizontal. 6.1.2.3 Sustained winds greater than 100 MPH AND Reports or indications of damage to vital equipment or structures.	
	6.2 High Lake / Low Forebay Water Level	6.2.1.1 Less than -11' forebay or pump bay level with one unit's CW pumps off. 6.2.2.1 Any flooding which precludes access to the site or areas of the plant.	6.2.2.2 Flooding as indicated by greater than 6" of water in the 8' elevation of the Turbine Building.	6.2.2.3 Greater than 2' of water in vital switchgear room OR Greater than 2' of water in auxiliary feedwater pump room	
	6.3 Toxic / Flammable Gas Intrusion	6.3.1.1 Near or on-site flammable or toxic gas release as reported to Shift Manager by plant personnel making visual observation.	6.3.1.2 Entry of toxic or flammable gas into a plant building atmosphere affecting operation or access.	6.3.1.3 Entry of toxic or flammable gas into a plant vital area affecting operation or personnel safety AND Reactor coolant temperature greater than 200°F.	
	6.4 Vehicle / Missile Impacts	6.4.1.1 Unusual aircraft activity over facility.	6.4.1.2 Aircraft crash in protected area (within the fence). 6.4.2.2 Missile impact from any source by visual observation of Operations Supervisor.	6.4.1.3 Aircraft crash affecting operability of two (2) trains of safety systems. 6.4.2.3 Any missile impact affecting operability of two (2) trains of safety systems.	
Fuel Handling/ISFSI Events	7.1 Fuel Handling Events		7.1.1.2 Report of possible damage to irradiated fuel combined with an alarm on any of the following radiation monitors: RE-211, Containment air particulate monitor. RE-212 Containment noble gas monitor. RE-221 Drumming Area Vent. Manipulator Area Monitor. Spent Fuel Bridge Area Monitor.		
	7.2 Irradiated Fuel Events		7.2.1.2 Indications of irradiated fuel uncovered.		
	7.3 ISFSI Events	7.3.1.1 A loaded spent fuel cask dropped or tipped.	7.3.1.2 Breach of a loaded spent fuel cask as indicated by a reading greater than 1000 mr/hr at 1 meter.		
Judgment		8.1.1.1 Any event which in the judgement of the Shift Manager or the Emergency Director could lead to, or has led to, a **potential degradation** of the level of safety of the plant.	8.1.1.2 Any event which in the judgement of the Shift Manager or the Emergency Director could cause or has caused **actual or potential substantial degradation** of the level of safety of the plant.	8.1.1.3 Any event which in the judgement of the Shift Manager or the Emergency Director could indicate **actual or likely major failures of plant functions** needed to protect the public. Any releases are not expected to result in exposures in excess of EPA PAGs.	8.1.1.4 Any event which in the judgement of the Shift Manager or the Emergency Director could lead to **actual or imminent core damage** and the potential for a large release of radioactive material (in excess of EPA PAGs) outside the site boundary.

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

REPLACED BY NEI 99-01 REV 4 SCHEME

Category: Fission Product Barriers

EAL 1.1.1.1

Sub-Category: None

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Reactor coolant sample activity greater than Technical Specification TS 3.4.16.

Basis:

This EAL is related to a Fission Product Barrier challenge. See Attachment C for additional information.

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and is a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant Technical Specifications (TSAC 3.4.16.B or TSAC 3.4.16.C has been entered).

Technical Specifications allow exceeding normal coolant activities for limited time periods (TSAC 3.4.16.A). This EAL does not apply while operating within these allowances.

Because RCS leakage and coolant activity are considered precursors to more serious events, and because they should be treated alike (each relating to a Fission Product Barrier) declaration shall be upon validation and shall **NOT** be delayed until Technical Specification's actions are taken.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 3b

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fission Product Barriers

EAL 1.1.1.2

Sub-Category: Loss of One Barrier

Emergency Classification: **ALERT**

Replaced by NEI 99-01 Rev 4 Scheme

Emergency Action Level:

Exceeding the LOSS threshold of either Fuel Clad OR Reactor Coolant System (RCS) barrier based on FPB Matrix (See Attachment C for thresholds).

Basis:

This Fission Product Barrier (FPB) EAL refers to exceeding the LOSS threshold of either the Fuel Cladding or Reactor Coolant System barrier by comparing plant conditions to the thresholds outlined in the FPB Matrix (Attachment C).

The FPB Matrix LOSS criteria indicate values at which either the Fuel Cladding or RCS barrier has been breached to the point that it no longer serves as an effective barrier to the travel of fission products. This value is not intended to represent total loss, however one of these two essential barriers is no longer serving its function. A substantial reduction in the level of safety at the plant exists, therefore an Alert classification is appropriate.

Loss of the Containment barrier (by itself) does not create an immediate transport of fission products as the Containment is designed to be a backup to the cladding and RCS barriers. Therefore, if only the Containment barrier is lost, it will be dealt with by Technical Specification action statements. However, if either the Fuel Cladding or RCS barrier is lost, the Containment barrier will be considered at the same level as these.

References:

NEI 97-03 Rev. 2

NUREG 0654, Appendix 1 Initiating Condition: Alert 1b,1c, 5

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Replaced By NEI 99-01 Rev 4 Scheme

Category: Fission Product Barriers

EAL 1.1.1.3

Sub-Category: Loss of Two Barriers

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Exceeding the LOSS threshold of any 2 fission product barriers based on FPB Matrix (See Attachment C for thresholds).

Basis:

This Fission Product Barrier (FPB) EAL refers to exceeding the LOSS threshold of any two of the three fission product barriers; fuel cladding, reactor coolant system, or containment by comparing plant conditions to the thresholds outlined in the FPB Matrix (Attachment C).

The third barrier must remain INTACT. If challenged or lost, a General Emergency exists.

The FPB Matrix LOSS criteria indicate values at which barriers have been breached to the point that they no longer serve as effective barriers to the travel of fission products. These values are not intended to represent total loss, however two important barriers are no longer serving their function. This represents a major failure in plant systems needed to protect the public, therefore a Site Emergency classification is appropriate.

References:

NEI 97-03

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fission Product Barriers

EAL 1.1.1.4

Sub-Category: Loss of Three Barriers

Emergency Classification: **GENERAL EMERGENCY**

REPLACED BY NEI 99-01 REV 4 SCHEME

Emergency Action Level:

Exceeding the LOSS threshold of any 2 fission product barriers AND exceeding the loss OR challenge threshold of the 3rd barrier based on the FPB Matrix (See Attachment C for thresholds).

Basis:

This Fission Product Barrier (FPB) EAL refers to exceeding the LOSS threshold of any TWO of the three fission product barriers; fuel cladding, reactor coolant system, or containment AND also exceeding EITHER the loss OR challenge threshold on the third barrier by comparing plant conditions to the thresholds outlined in the FPB Matrix (Attachment C).

The FPB Matrix LOSS criteria indicate values at which barriers have been breached to the point that they no longer serve as effective barriers to the travel of fission products. These values are not intended to represent total loss, however the barriers are no longer serving their function. The loss of two and a loss or challenge of the third available barrier represents major failures to plant systems needed to protect the public with the actual or potential release of significant amounts of radioactive materials offsite, therefore a General Emergency classification is appropriate.

References:

NEI 97-03 Rev.2

NUREG 0654, Appendix 1 Initiating Condition: General Emergency 2

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fission Product Barriers

EAL 1.1.2.1

Sub-Category: None

Emergency Classification : **UNUSUAL EVENT**

Emergency Action Level:

Failed fuel monitor [1(2) RE-109] reading greater than 120 mRem/hr, or 2 of 3 containment high range monitors read greater than 1000 Rem/hr.

Basis:

This EAL is related to a Fission Product Barrier challenge. See Attachment C for additional information.

Other indications should accompany this indication, such as increased radiation on RE-106 or on hand-held instruments.

Elevated reactor coolant activity as indicated by the failed fuel monitor [1(2) RE-109] represents a potential degradation in the level of safety of the plant and is a potential precursor of more serious problems. This EAL addresses failed fuel monitor readings exceeding approximately 0.1% fuel clad failures.

Because RCS leakage and coolant activity are considered precursors to more serious events, and because they should be treated alike (each relating to a Fission Product Barrier) declaration shall be upon validation and shall **NOT** be delayed until Technical Specification's actions are taken.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 3c

REPLACED BY NEI 99-01 Rev Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fission Product Barriers

EAL 1.1.2.2

Sub-Category: Loss of One Barrier

Replaced By NEI 99-01 Rev 4 Scheme

Emergency Classification: **ALERT**

Emergency Action Level:

Unisolable steam line break outside containment with greater than 10 gpm , but less than 50 gpm, primary to secondary leakage.

Basis:

This EAL reflects a unique Initiating Condition from NUREG-0654. It does not meet the loss of one barrier criteria from Attachment C (Fission Product Barrier matrix), yet will be classified as an Alert.

Because an unisolable steam line break is evaluated under the *Containment* section of the Fission Product Barrier matrix, it would not result in an Alert by itself. Because the primary to secondary leakage rate (10 gpm) is less than the LOSS criteria for RCS, it would not result in an Alert. The 10 gpm *does* meet the CHALLENGE criteria therefore is an Unusual Event. However, there is no logic in the FPB matrix for combinations of LOSS of Containment with CHALLENGE of another barrier.

Due to the unique, specific criteria of NUREG-0654, Appendix 1 criteria, this EAL covers the unique condition of an unisolable steam line break, combined with a small primary to secondary leak.

If the steam line can be isolated, no emergency is warranted. If the steam line cannot be isolated, and the other Fission Product Barriers are INTACT (No leakage, or leakage below 10 gpm) then no emergency is applicable UNLESS the SM determines a potential degradation in the level of safety.

If the steam line cannot be isolated AND primary to secondary leakage is greater than 10 gpm, but less than 50 gpm, then this EAL applies and an Alert must be declared.

If the primary to secondary leak rate exceeds 50 gpm, then the LOSS criteria for RCS Fission Product Barrier has been met. This would constitute LOSS of two barriers, and would be a Site Emergency on EAL 1.1.1.3.

References:

NEI 97-03 Rev. 2

NUREG 0654, Appendix 1 Initiating Condition: Alert 4

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fission Product Barriers

EAL 1.1.3.1

Sub-Category: None

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Primary to secondary leakage greater than Technical Specification Reference TS 3.4.13.d (500 gallons per day through any one steam generator).

Basis:

This EAL is related to a Fission Product Barrier challenge. See Attachment C for additional information.

Leakage from the RCS in excess of Technical Specifications is considered by the NRC to be a precursor to more serious events. Therefore, an Unusual Event must be declared even if Technical Specification actions are taken.

Because RCS leakage and coolant activity are considered precursors to more serious events, and because they should be treated alike (each relating to a Fission Product Barrier) declaration shall be upon validation and shall **NOT** be delayed until Technical Specification's actions are taken.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 5

Replaced by NEI 99-01 Rev 4 scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fission Product Barriers

EAL 1.1.4.1

Sub-Category: None

Emergency Classification: **UNUSUAL EVENT**

Replaced By NEI 99-01 Rev 4 scheme

Emergency Action Level:

Unisolable primary system leakage greater than Technical Specification Reference TS 3.4.13.c (10 gallons per minute).

Basis:

This EAL is related to a Fission Product Barrier challenge. See Attachment C for additional information.

Leakage from RCS in excess of Technical Specifications which cannot be isolated is considered by the NRC to be a precursor to more serious events. Therefore, an Unusual Event must be declared even if Technical Specification actions are taken.

Because RCS leakage and coolant activity are considered precursors to more serious events, and because they should be treated alike (each relating to a Fission Product Barrier) declaration shall be upon validation and shall **NOT** be delayed until Technical Specification's actions are taken.

References:

PBNP Technical Specifications

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 5

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fission Product Barriers

EAL 1.1.5.1

Sub-Category: None

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Excess RCS cooldown or cold overpressurization of the RCS (ST-4 Integrity Orange path)

Basis:

The following conditions meet ST-4 Integrity - Orange Path criteria. A challenge to the RCS barrier is present due to excessive cooldown or cold overpressurization as indicated below:

Decrease in temperature in either cold leg greater than 100°F in the last 60 minutes AND temperature in either cold leg less than 315°F.

OR

Temperature in either cold leg less than 315°F and RCS pressure greater than 425 psig.

Any actual loss of RCS barrier warrants declaration of an Alert per the FPB matrix, Attachment C.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 17

REPLACED BY NEI 99-01 REV SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.1.1.2

Sub-Category: Failure to Trip

Emergency Classification: **ALERT**

Emergency Action Level:

Failure of the reactor protection system(automatic or manual) to initiate and complete a trip which brings the reactor subcritical.

Basis:

The reactor protection system may be actuated either by automatic means (exceeding pre-determined thresholds which result in trip signals) or by operator action (manual trip).

The failure of EITHER of these means to cause a trip with subsequent subcriticality meets this EAL (an Alert).

If BOTH these means <u>AND</u> all other means from the Control Room fail, see EAL 2.1.1.3 (a Site Emergency).

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 11

Replaced By NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.1.1.3

Sub-Category: Failure to Trip

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Failure to rapidly bring the reactor subcritical from the Control Room. (ST-1 Subcriticality Red Path)

Basis:

CSFST Subcriticality - RED path is entered based on failure of power range indication (N-41, N-42, N-43, N-44) to decrease below 5% following a reactor trip. This EAL addresses any manual trip or automatic trip signal followed by a manual trip or other Control Room actions which fail to rapidly shut down the reactor.

If any actions must be taken outside the Control Room to effect a reactor trip this EAL is also met.

This condition indicates failure of both the automatic and manual protection systems to trip the reactor, to an extent that emergency boration is required: or actions are needed outside the Control Room to trip the reactor. The failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to fuel clad and RCS integrity and thus warrants declaration of a Site Emergency.

This EAL is synonymous with entry into CSP S-1.

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 9

Replaced By NEW EAL Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.2.1.1

Sub-Category: Technical Specification Requirements

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Failure to reach Technical Specification required operating mode or condition within the specified time limit of the LCO action statement.

Basis:

Limiting Conditions of Operation (LCOs) action statements require the plant to be brought to a required condition (often shutdown) 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 Technical Specification requires a four hour report under 10 CFR 50.72 (b) non-emergency events. The plant remains within its evaluated safety envelope while changing conditions or being shut down so long as it is accomplished within the completion time for the required action in the Technical Specifications.

An immediate Unusual Event is required when the plant is not brought to the required operating mode or condition within the allowable action statement time of the Technical Specifications. Declaration of an Unusual Event 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.

If a Notice of Enforcement Discretion (NOED) is approved by the NRC prior to the LCO action statement time expiration an emergency need not be declared.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 15

Replaced by NUMARC Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.3.1.1

Sub-Category: Loss of Indications/Communications

Replaced by NEI 99-01 Rev 4 Scheme

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Unplanned loss of most (approximately 75%) safety system annunciators or indications on Control Room Panels for greater than 15 minutes ***AND*** *increased monitoring is required for safe plant operation.*

Basis:

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment. Recognition of the normal availability of computer based indication equipment is considered.

"Unplanned" loss of annunciators or indicators excludes scheduled maintenance and testing activities, which should not disable such large portions of the system(s).

It is not intended that personnel perform a count of the instrumentation or annunciation lost but use the judgment of the SM as the threshold for determining the severity of the plant condition. The increased monitoring portion of this EAL is met if the SM determines that additional personnel are required to provide increased monitoring of system operation to safely operate the plant.

It is recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible 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 safety system indicators should remain a function of that specific system or component operability status, and is addressed by the specific Technical Specifications.

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

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

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 14

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.3.1.2

Sub-Category: Loss of Indications/Communications

Emergency Classification: **ALERT**

REPLACED BY NEI 99-01

Emergency Action Level:

Unplanned loss of most (approximately 75%) safety system annunciators or indications on Control Room Panels for greater than 15 minutes
AND
Increased monitoring is required for safe plant operation
AND either:
A significant plant transient is in progress
OR
PPCS is unavailable.

Basis:

This EAL recognizes 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 normal availability of computer based indication equipment is also considered.

"Unplanned" loss of annunciators or indicators excludes scheduled maintenance and testing activities, which should not disable such large portions of the system(s).

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

It is not intended that personnel perform a count of the instrumentation or annunciation lost but the use the judgment of the SM as the threshold for determining the severity of the plant conditions. The increased monitoring portion of this EAL is met if the SM determines that additional personnel are required to provide increased monitoring of system operation to safely operate the plant.

It is recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible 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 and is addressed by the specific Technical Specifications.

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

"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 ramps of 10% or greater.

If both a major portion of the annunciation system and all computer monitoring is unavailable to the extent that additional personnel are required to monitor indications, the Alert is required. If the operating crew cannot monitor the transient in progress this will be escalated to a Site Emergency.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 14

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.3.1.3

Sub-Category: Loss of Indications/Communications

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Unplanned loss of most (approx. 75%) safety system annunciators or indications on Control Room Panels. *AND* *Loss of ability to monitor critical safety function status* *AND* *A significant plant transient in progress.*

Basis:

This EAL recognizes the INABILITY of the Control Room staff to monitor the plant response to a transient. A Site 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 ramps of 10% or greater.

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

"Unplanned" loss of annunciators or indicators excludes scheduled maintenance and testing activities, which should not disable such large portions of the system(s).

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 12

REPLACED BY NEI 99-01 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.3.2.1

Sub-Category: Loss of Indications/Communications

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Loss of all communications capability affecting the ability to either:
Perform routine operations
OR
Notify offsite agencies or personnel.

Basis:

The purpose of this EAL is to recognize a loss of communications capability that EITHER defeats the plant operation's or staff's ability to perform routine tasks necessary for plant operations OR the ability to communicate problems with offsite authorities. The loss of offsite communications ability anticipated by this EAL is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The onsite communications loss must encompass the loss of all means of routine communications (i.e., plant telephone system, Gai-tronics page system, portable radios).

The offsite communications loss must encompass the loss of all means of communications with offsite authorities. This should include Emergency Notification System (ENS) for NRC, Microwave lines, and radio. This EAL is also met when extraordinary means are being utilized to make communications possible (relaying of information from radio transmissions, individuals being sent to offsite locations, etc.).

Procedure DCS 2.1.1 describes lesser communications losses which must be reported to the NRC within eight hours.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 11

Replaced by NMC-01 Rev Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.4.1.2

Sub-Category: Degradation of Safety System Performance

Emergency Classification: **ALERT**

Emergency Action Level:

Inability to maintain reactor coolant temperature less than or equal to 200°F.

Basis:

This EAL addresses complete loss of functions required for core cooling during refueling and cold shutdown modes. Escalation to Site Emergency or General Emergency would be through other EALs.

An uncontrollable reactor coolant temperature increase that approaches or exceeds the cold shutdown technical specification limit warrants declaration of an Alert. The concern of this EAL is the loss of control resulting in the loss of ability to maintain the plant in cold shutdown which is defined by reactor coolant temperature.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 10

REPLACED BY NEI 99-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.4.1.3

Sub-Category: Degradation of Safety System Performance

Emergency Classification: **SITE EMERGENCY**

Emergency Action Levels:

Primary to secondary leakage greater than 400 gallons per minute *AND* *Inability to power BOTH buses A-05 AND A-06 from offsite sources.*

Basis:

400 gpm is also the expected output from a single SI pump @ 1400 psia RCS pressure. (See DBD-09).

Loss of offsite power combined with an RCS leak (from Primary to Secondary) of this magnitude constitute several major challenges to the protection of the public:

1. Operating on diesel generators.
2. Leak (rupture) near the capacity of a single Safety Injection pump.
3. Transport of nay fission products from Primary to Secondary.

Therefore, major plant functions needed for the protection of the public have been affected. A Site Emergency is warranted.

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 3

Replaced by NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.5.1.1

Sub-Category: Reactivity Transient

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Uncontrolled Rod Withdrawal (FSAR 14.1.1 and 14.1.2)

Basis:

A malfunction which results in an uncontrolled withdrawal of control rod(s) is a reactivity transient which indicates a potential degradation of the level of safety of the plant. This condition warrants an Unusual Event Classification.

Uncontrolled is defined as unwarranted rod motion that cannot be prevented by operator action (i.e., going to manual).

The Unusual Event classification is warranted if a reactor trip is required to stop rod motion.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 15.

FSAR 14.1.1, Uncontrolled Rod Withdrawal from Subcritical.

FSAR 14.1.2, Uncontrolled Rod Withdrawal at Power.

REPLACED BY NEI 99-01 Rev 2 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: System Malfunctions

EAL 2.6.1.4

Sub-Category: Feedwater Transient

Emergency Classification: **GENERAL EMERGENCY**

Emergency Action Level:

Transient Initiated By Loss of Feedwater, followed by loss of auxiliary feedwater for ***greater than*** *1 hour. As indicated by* ***All*** *of the following:*

1. *Decreasing SG Levels*
   "A" SG [LI-461, LI-462, LI-463]
   "B" SG [LI-471, LI-472, LI-473]
2. *No auxiliary feedwater flow*
   [FI-4002, FI-4007, FI-4014, FI-4036, FI-4037]

Basis:

This EAL assures that in the event of a prolonged total loss of feedwater, timely recognition of the loss of heat sink occurs.

Therefore, this condition is indicative of actual or imminent substantial core degradation with potential adverse consequences on the public health and safety. A GENERAL EMERGENCY is warranted.

References:

NUREG 0654, Appendix 1 Initiating Condition: General Emergency 5b

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Loss of Electrical Power

EAL 3.1.1.1

Sub-Category: Loss of Vital AC Power

Emergency Classification: **UNUSUAL EVENT**

REPLACED BY NEI 99-01 REV 4 SCHEME

Emergency Action Level:

*Loss of all **offsite** AC capability to vital buses as indicated by the inability to power BOTH buses A-05 AND A-06 of a given unit from offsite sources for greater than 15 minutes.*

Basis:

Prolonged loss of offsite 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 (station blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Therefore, this condition (which is indicative of degraded conditions, but with no adverse consequences on the public health and safety) is classified as an UNUSUAL EVENT.

If primary to secondary leakage also exists, see EAL 2.4.1.3.

References:

FSAR Section 8, Electrical Systems

DBD-22, 4160 VAC System

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 7a

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Loss of Electrical Power

EAL 3.1.1.2

Sub-Category: Loss of Vital AC Power

Emergency Classification: **ALERT**

Emergency Action Level:

Loss of ***all*** *safeguard bus AC power of a given unit as indicated by the inability to power BOTH buses A-05 AND A-06, OR B-03 AND B-04.* *AND* *Loss is for less than 15 minutes.*

Basis:

Loss of all AC power safeguards buses compromises critical plant safety functions including RHR, ECCS, containment heat removal, and maintaining the ultimate heat sink. Prolonged loss of all AC power safeguards buses may result in uncovering the core and loss of containment integrity, thus this event can escalate to a General Emergency. The site blackout coping analysis assumes that AC power can be restored in one hour.

This condition is entered when there are indications of a total loss of power to the safeguards buses A-05 and A-06 OR B-03 and B-04 from any source (on or off-site) for less than 15 minutes.

This condition is indicative of actual or potential substantial degradation to plant systems with possible adverse consequences on the public health and safety. An ALERT is warranted and must be declared.

This EAL escalates to a SITE EMERGENCY if loss of AC power continues for greater than 15 minutes.

References:

FSAR Section 8, Electrical Systems

DBD-22, 4160 VAC System

NUREG 0654, Appendix 1 Initiating Condition: Alert 7

Replaced By NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Loss of Electrical Power

EAL 3.1.1.3

Sub-Category: Loss of Vital AC Power

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

*Loss of **all** safeguard bus AC power of a given unit as indicated by the inability to power BOTH buses A-05 AND A-06, OR B-03 AND B-04.* *<u>AND</u>* *Loss is for greater than 15 minutes.*

Basis:

Loss of all AC power safeguards buses compromises critical plant safety functions including RHR, ECCS, containment heat removal, and maintaining the ultimate heat sink. Prolonged loss of all AC power safeguards buses may result in the uncovering core and loss of containment integrity, thus this event can escalate to a General Emergency. The site blackout coping analysis assumes that AC power can be restored in one hour.

This condition is entered when there are indications of a total loss of power to the safeguards buses A-05 and A-06 OR B-03 and B-04 from any source (on or off-site) for more than 15 minutes.

Therefore, this condition (which is indicative of serious plant system conditions with adverse consequences on the public health and safety) is classified as a SITE EMERGENCY.

References:

FSAR Section 8, Electrical Systems

DBD-22, 4160 VAC System

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 6

REPLACED BY NEI [illegible] REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Loss of Electrical Power

EAL 3.1.1.4

Sub-Category: Loss of Vital AC Power

Emergency Classification: **GENERAL EMERGENCY**

REPLACED BY NEW 04-01 REV SCHEME

Emergency Action Level:

*Loss of **all** safeguard bus AC power of a given unit as indicated by the inability to power BOTH buses A-05 AND A-06, OR B-03 AND B-04.* ***AND*** *Loss is greater than 15 minutes.* ***AND*** *Both narrow range S/G level less than [51%] 29% **AND** total feedwater flow to S/Gs less than 200 gpm. (ST-3 Heat Sink Red path)*

Basis:

Loss of all AC power safeguards buses compromises critical plant safety functions including RHR, ECCS, containment heat removal, and maintaining the ultimate heat sink. Prolonged loss of all AC power safeguards buses may result in the uncovering core and loss of containment integrity, thus this event can escalate to a General Emergency. The site blackout coping analysis assumes that AC power can be restored in one hour.

This EAL assures that in the event of a prolonged station blackout, timely recognition of the loss of heat sink occurs.

Therefore, this condition is indicative of grave plant conditions with potential adverse consequences on the public health and safety. A GENERAL EMERGENCY is warranted and must be declared.

References:

FSAR Section 8, Electrical Systems

DBD-22, 4160 VAC System

NUREG 0654, Appendix 1 Initiating Condition: General Emergency 5d

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

REPLACED BY NEI 99-01 REV 4 SCHEME

Category: Loss of Electrical Power

EAL 3.1.2.1

Sub-Category: Loss of Vital AC Power

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Loss of all ***onsite*** *AC power capability to power BOTH buses A-05 AND A-06 of a given unit from onsite sources (GO1 through GO4) for greater than 15 minutes.*

Basis:

Loss of onsite safety related AC power sources 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 (station blackout). Therefore, an Unusual Event is warranted and must be declared. Fifteen minutes was selected as a threshold to exclude transient losses.

This condition is entered when there are indications of the unavailability of all the emergency diesel generators (GO1 through GO4) or that none of these sources can be aligned to either A-05 or A-06 for greater than 15 minutes.

Therefore, this condition (which is indicative of degraded conditions, but with no adverse consequences on the public health and safety) is classified as an Unusual Event.

References:

FSAR Section 8, Electrical Systems

DBD-22, 4160 VAC System

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 7b

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Loss of Electrical Power

EAL 3.2.1.2

Sub-Category: Loss of Vital DC Power

Emergency Classification: **ALERT**

Emergency Action Level:

Loss of all vital DC power as indicated by less than 105 vdc on all station battery buses (D01, D02, D03, D04) for less than 15 minutes.

Basis:

Loss of all vital DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power may result in uncovering the core and loss of containment integrity.

Loss of DC power to any AC bus creates the following conditions:

1. Associated breakers cannot be electrically opened or closed remotely or locally;

2. Electrical protection/interlock tripping of associated breakers is rendered inoperable including undervoltage stripping. The one exception is the 480 V individual breaker overloads which remain operable;

3. All associated breaker positions remain AS IS.

Loss of all vital onsite DC power may also be indicated by an "Annunciator Power Failure" alarm.

This EAL escalates to a SITE EMERGENCY if the power loss continues for greater than 15 minutes.

References:

FSAR Section 8, Electrical Systems

DBD-19, 125 VDC System

NUREG 0654, Appendix 1 Initiating Condition: Alert 8

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Loss of Electrical Power **EAL 3.2.1.3**

Sub-Category: Loss of Vital DC Power

Emergency Classification: **SITE EMERGENCY**

Replaced By NEI 99-01 Rev 4 Scheme

Emergency Action Level:

Loss of all vital DC power as indicated by less than 105 vdc on all station battery buses (D01, D02, D03, D04) for greater than 15 minutes.

Basis:

Loss of all vital DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power may result in uncovering the core and loss of containment integrity.

Loss of DC power to any AC bus creates the following conditions:

1. Associated breakers cannot be electrically opened or closed remotely or locally;
2. Electrical protection/interlock tripping of associated breakers is rendered inoperable including undervoltage stripping. The one exception is the 480 V individual breaker overloads which remain operable.
3. All associated breaker positions remain AS IS.

Loss of all vital onsite DC power may also be indicated by an "Annunciator Power Failure" alarm.

This condition (which is indicative of possible loss of control of the reactor coolant and containment barriers, with possible adverse consequences on the public health and safety) is classified as a SITE EMERGENCY.

References:

FSAR Section 8, Electrical Systems

DBD-19, 125 VDC System

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 7

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Radiological Conditions

EAL 4.1.1.1

Sub-Category: Off-site Radiological Release

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Vent radiation reading(s) exceed the high alarm setpoints for greater than 60 minutes,
OR
Liquid release in excess of high alarm setpoints which cannot be isolated.

Replaced by NEI 99-01 Rev 4 Scheme

Vent Radiation High Alarm Setpoints 9/99	Reference RMSARB for current setpoint values	
1 RE 212	2.73 E-4 uCi/cc	if purging, 1.62E-2 if forced vent.
1 RE 215	2.71 E+0 uCi/cc	
RE 214	1.02 E–4 uCi/cc	
RE 221	1.58 E-4 uCi/cc	
RE 224	2.09 E-3 uCi/cc	
RE 225	1.36 E+0 uCi/cc	
2 RE 212	1.78 E-4 uCi/cc	if purging, 1.82E-2 if forced vent.
2 RE 215	2.71 E + 0 uCi/cc	

Liquid Release Limits		
Service Water Discharge 1(2) RE-229 High Alarm AND:		Waste Water Effluent RE-230 High Alarm AND:
1 Circ. Water pump **AND:**	**Release Limit** (uCi/cc):	1 Circ. Water pump **Release Limit** (uCi/cc): 3.70 E-4
2 Service Water pumps	4.12 E-5	
3 Service Water pumps	3.27 E-5	
4 Service Water pumps	3.03 E-5	
5 Service Water pumps	2.87 E-5	
6 Service Water pumps	2.78 E-5	
2 Circ. Water pump **AND:**	**Release Limit** (uCi/cc):	2 Circ. Water pump **Release Limit** (uCi/cc): 6.29 E-4
2 Service Water pumps	7.00 E-5	
3 Service Water pumps	5.56 E-5	
4 Service Water pumps	5.15 E-5	
5 Service Water pumps	4.88 E-5	
6 Service Water pumps	4.73 E-5	

Reference:
C.H. Onesti to G.J. Maxfield, 11/17/92, RE-229 and RE-230 Alarm Setpoints, NPM 92-1035.

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Basis:

Unplanned airborne releases in excess of the site technical specifications, that cannot be reduced to within technical specifications within 60 minutes, represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not controlled to within Technical Specification limits within 60 minutes.

Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 2 times Technical Specifications for 30 minutes, but which is terminated, does not exceed this EAL. However, the SM 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.

Likewise, liquid release values (which would result in very low integrated dose) are not the primary concern. Rather, the fact that the release cannot be isolated represents a potential degradation in the level of safety.

References:

STPT 13.4, Effluent Monitors

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 2

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Radiological Conditions

EAL 4.1.1.2

Sub-Category: Off-site Radiological Release

Emergency Classification: **ALERT**

Emergency Action Level:

Vent radiation readings exceed ***ten times*** *the high alarm setpoints for greater than 15 minutes.*
OR
Liquid release in excess of ***ten times*** *high alarm setpoint which cannot be isolated.*

10 times Vent Radiation High Alarm Setpoints 9/99	Reference RMSARB for current setpoint values	
1 RE 212	2.73 E-3 uCi/cc	if purging, 1.62E-1 if forced vent
1 RE 215	2.71 E+1 uCi/cc	
RE 214	1.02 E–3 uCi/cc	
RE 221	1.58 E-3 uCi/cc	
RE 224	2.09 E-2 uCi/cc	
RE 225	1.36 E+1 uCi/cc	
2 RE 212	1.78 E-3 uCi/cc	if purging, 1.82E-1 if forced vent
2 RE 215	2.71 E+1 uCi/cc	

Liquid Release Limits Service Water Discharge 1(2) RE-229 High Alarm AND:		Waste Water Effluent RE-230 High Alarm AND:
1 Circ. Water pump AND:	Ten times Release Limit (uCi/cc):	1 Circ. Water pump Ten times Release Limit (uCi/cc): 3.70 E-3
2 Service Water pumps	4.12 E-4	
3 Service Water pumps	3.27 E-4	
4 Service Water pumps	3.03 E-4	
5 Service Water pumps	2.87 E-4	
6 Service Water pumps	2.78 E-4	
2 Circ. Water pump **AND:**	**Ten times Release Limit** (uCi/cc):	**2** Circ. Water pump **Ten times Release Limit** (uCi/cc): 6.29 E-3
2 Service Water pumps	7.00 E-4	
3 Service Water pumps	5.56 E-4	
4 Service Water pumps	5.15 E-4	
5 Service Water pumps	4.88 E-4	
6 Service Water pumps	4.73 E-4	

Reference:
C.H. Onesti to G.J. Maxfield, 11/17/92, RE-229 and RE-230 Alarm Setpoints, NPM 92-1035.

Replaced by NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Basis:

Release rates in excess of ten times technical specifications which continue for 15 minutes or longer represent a serious situation. Ideally, most releases will begin small, then increase, hence will progress through the Unusual Event classification, allowing time to stop or mitigate them. Assuming this is the case, significant time has passed during which attempts to reduce or terminate the release have failed. Therefore the required release duration for meeting this EAL was reduced to 15 minutes in recognition of the increased severity.

The final integrated dose (which is still expected to be low at these release rates) is not the primary concern here; it is the degradation in plant control implied by the fact that the release cannot be controlled.

Likewise, liquid release values (which would result in very low integrated dose) are not the primary concern. Rather, the fact that the release cannot be isolated represents a potential degradation in the level of safety.

References:

STPT 13.4, Effluent Monitors

NUREG 0654, Appendix 1 Initiating Condition: Alert 15

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Radiological Conditions

EAL 4.1.1.3

Sub-Category: Off-site Radiological Release

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

a. *Effluent monitors detect levels corresponding to either:*

(1) 0.1 Rem Total Effective Dose Equivalent (TEDE).

(2) 0.5 Rem thyroid Committed Dose Equivalent (CDE)

at the site boundary under actual meteorological conditions.

b. *Either of the above doses measured in the environs.*

c. *Either of the above doses projected based on plant parameters.*

REPLACED BY REV 49-01

Basis:

The 0.1 rem TEDE is based on the 10 CFR 20 annual average population exposure. This value also provides a desirable gradient (one order of magnitude) between the Site Emergency and General Emergency classes. It is deemed that exposures less than this limit are not consistent with the Site Emergency class description. The 0.5 Rem CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for whole body to thyroid.

Dose projection can be based on values obtained from effluent monitors, direct measurements taken in the environment, or any other appropriate plant parameters.

Integrated doses are not monitored in real-time but are projected. In establishing the duration used for the projection, care should be exercised to ensure the time estimates are realistic. If no educated guess can be made regarding estimated duration, the default (4 hours) shall be used.

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 13a

EPPOS1, on acceptable Deviation from Appendix 1 of NUREG 0654

EPA 400 Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

REPLACED BY NEI 99-01 Rev 4 Scheme

Category: Radiological Conditions

EAL 4.1.1.4

Sub-Category: Off-site Radiological Release

Emergency Classification: **GENERAL EMERGENCY**

Emergency Action Level:

a. *Effluent monitors detect levels corresponding to either:*

(1) 1 Rem Total Effective Dose Equivalent (TEDE).

(2) 5 Rem thyroid Committed Dose Equivalent (CDE)

at the site boundary under actual meteorological conditions.

b. *Either of the above doses measured in the environs.*

c. *Either of the above doses projected based on plant parameters.*

Basis:

The 1 REM TEDE and the 5 REM CDE thyroid integrated doses are based on the EPA protective action guidance which indicates that public protective actions are indicated. This is consistent with the emergency class description for a General Emergency.

Dose projection can be based on values obtained form effluent monitors, direct measurements taken in the environment, or from any other appropriate plant parameters.

Integrated doses are not monitored in real-time but are projected. In establishing the duration used for the projection, care should be exercised to ensure the time estimates are realistic. If no educated guess can be made regarding estimated duration, the default (4 hours) shall be used.

References:

NUREG 0654, Appendix 1 Initiating Condition: General Emergency 1a

EEPOS1, EPPOS on Acceptable Deviation from Appendix 1 of NUREG 0654

EPA 400 Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Radiological Conditions

EAL 4.2.1.2

Sub-Category: In-Plant Radiological Conditions

Emergency Classification: **ALERT**

Emergency Action Level:

Loss of control of radioactive material resulting in area radiation exceeding 1000X normal (or expected) levels within the Protected Area. Normal may be determined by trend recorder or other relevant data.

Basis:

By themselves indications of increased levels of radiation would only meet the Unusual Event class description (potential degradation in the level of safety). However, there is no specific Unusual Event EAL on increased radiation. This would be a judgment call by the SM. However, when increased radiation of *this* magnitude (1000x) is *combined* with "loss of control" a higher classification is warranted. Non-essential personnel should be assembled to ensure their safety. Additional manpower or other resources may be needed. The ALERT classification is appropriate.

The operative phrase in this EAL is "loss of control". Combined with this is the phrase "or expected levels". For most plant evolutions increases of radiation can be estimated, most within a factor of 1000. If, in the judgment of those concerned, control has been lost AND radiation levels increase beyond 1000X normal or expected levels, this EAL is met.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 6

Replaced By [illegible] Rev Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Replaced by New EAL Scheme

Category: Internal Events

EAL 5.1.1.1

Sub-Category: Security Threats

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Bomb, credible bomb threat, indication of sabotage, or attempted entry into the Protected Area by a hostile force.

Basis:

This EAL is based on the PBNP Security Plan/ISFSI Security Plan. An actual bomb, credible bomb threat, act of sabotage, or attempted entry into the Protected area by a hostile force indicates a potential degradation in the level of safety at the plant. Therefore an Unusual Event classification is warranted.

The Protected Area Physical Barrier is defined in the Security Plan/ISFSI Plan.

A bomb discovered in or near a Plant Vital Area which could affect Safety-Related Functions would result in escalation of the emergency classification. An actual explosion (of a bomb or other source) would be classified based on EALs 5.3.1.1 through 5.3.1.3 depending upon its effects.

Security events that do not represent at least a potential degradation in the level of plant safety are reported under either 10 CFR 73.71 or 10 CFR 50.72 and do not require implementation of the Emergency Plan. Accidental, non-hostile entry, although reportable as a security event, does not warrant declaration of an emergency. The operative consideration is 'intent'. If no malicious intent is determined the EAL does not apply.

References:

SSCP - Security and Safeguards Contingency Plan

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 12

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.1.1.2

Sub-Category: Security Threats

Emergency Classification: **ALERT**

Emergency Action Level:

Intrusion into the Protected Area by a hostile force.

Basis:

For the purposes of this EAL, the intrusion into the Protected Area can be considered a significant security threat. An Alert classification is warranted. If entry is attempted, but not gained by a hostile force see the Unusual Event EAL.

The Protected Area Physical Barrier is defined in the Security Plan. Note: The Independent Spent Fuel Storage Installation (ISFSI) is a separate Protected Area.

Intrusion into a Plant Vital Area escalates this event to a Site Emergency.

Security events that do not represent at least a potential degradation in the level of plant safety are reported under either 10 CFR 73.71 or 10 CFR 50.72 and do not require implementation of the Emergency Plan. Accidental, non-hostile entry, although reportable as a security event, does not warrant declaration of an emergency. The operative consideration is 'intent'. If no malicious intent is determined the EAL does not apply.

References:

SSCP - Security and Safeguards Contingency Plan

NUREG 0654, Appendix 1 Initiating Condition: Alert 16

Replaced by NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.1.1.3

Sub-Category: Security Threats

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Intrusion into a plant Vital Area by hostile force.

Basis:

Hostile takeover of Vital Areas could lead to loss of physical control of the plant. Therefore a Site Emergency classification is warranted. The Plant Vital Areas are defined in the Security Plan.

Security events that do not represent at least a potential degradation in the level of plant safety are reported under either 10 CFR 73.71 or 10 CFR 50.72 and do not require implementation of the Emergency Plan. Accidental, non-hostile entry, although reportable as a security event, does not warrant declaration of an emergency. The operative consideration is 'intent'. If no malicious intent is determined the EAL does not apply.

References:

SSCP - Security and Safeguards Contingency Plan

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 14

Replaced By NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.1.1.4

Sub-Category: Security Threats

Emergency Classification: **GENERAL EMERGENCY**

Emergency Action Level:

A Security Event which results in either: *Loss of physical control of the Control Room*   ***OR***   *Loss of remote shutdown capability.*

Replaced By NEI 99-01 Rev 4 Scheme

Basis:

This EAL encompasses conditions under which unauthorized personnel have taken physical control of vital areas required to reach and maintain safe shutdown, with the potential that the intruders can cause a significant event with damage to plant systems, damage to the core, and ultimately a release of large amounts of radioactivity.

References:

SSCP - Security and Safeguards Contingency Plan

AOP-10A, Safe Shutdown - Local Control

NUREG 0654, Appendix 1 Initiating Condition: General Emergency 5

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.2.1.2

Sub-Category: Control Room Habitability

Emergency Classification: **ALERT**

Emergency Action Level:

Evacuation of the Control Room has been initiated with control of shutdown systems established from local stations.

REPLACED BY NEI 99-01 REV 4 SCHEME

Basis:

AOP-10A directs shutdown activities performed outside the Control Room.

This EAL does not imply that all actions associated with Alternate Shutdown shall be completed in order to avoid the higher EAL pertaining to Control Room evacuation (EAL 5.2.1.3). If the reactor successfully trips, if level, pressure, temperature, etc., are being controlled, and no impediments to the associated Shutdown activities are being encountered, this emergency classification is appropriate. If impediments are being encountered in completing critical Shutdown functions, and more than 15 minutes expire, EAL 5.2.1.3 is met.

Located within the Control Room are the controls, indications, annunciators, and communications equipment necessary for the safe operation of the plant. The ability to assess and control plant conditions and abnormal situations is significantly degraded without access to the Control Room.

With the Control Room evacuated, additional support, monitoring, and direction through the resources of the TSC and/or other emergency facilities is assumed to be necessary - therefore, the declaration of an Alert is appropriate and required.

References:

AOP-10A, Safe Shutdown - Local Control

NUREG 0654, Appendix 1 Initiating Condition: Alert 20

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Replaced by NEI Rev Scheme

Category: Internal Events

EAL 5.2.1.3

Sub-Category: Control Room Habitability

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Evacuation of the Control Room without establishment of plant control from remote shutdown stations within approximately 15 minutes.

Basis:

Located within the Control Room ~~are~~ the controls, indications, annunciators, and communications equipment necessary for the safe operation of the plant. The ability to assess and control plant conditions and abnormal situations ~~is~~ significantly degraded without access to the Control Room.

Once the Control Room is evacuated, if control is not established from remote shutdown stations within a reasonable amount of time (approximately 15 minutes), a significant threat to multiple fission product barriers exists should a plant transient or other emergency condition occur. If plant control cannot be established within this time frame, declaration of a Site Emergency is warranted due to extended lack of control of the plant.

Escalation to a higher classification, if appropriate, will be based on system malfunctions, fission product barrier degradation, radiation levels, or Emergency Director judgment.

References:

AOP-10A, Safe Shutdown - Local Control

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 18

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

REPLACED BY NEI 99-01 REV 4 SCHEME

Category: Internal Events

EAL 5.3.1.1

Sub-Category: Fire / Explosion

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Near or on-site explosion as reported to Shift Manager by plant personnel making visual observation.

Basis:

No attempt is made in this EAL to assess the magnitude of damage. Reports of any explosion is sufficient for declaration.

On-site is defined as the exclusion area which is the area within the site boundary surrounding PBNP in which the plant personnel have the authority to determine all activities including exclusion or removal of personnel and property from the area. At PBNP the outer boundary of the exclusion area is coincident with the site boundary. (Reference Appendix C of Emergency Plan).

As used here, an explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized equipment imparting significant energy to nearby structures and materials. If the explosion damages Safety Systems the event escalates to an Alert or Site Emergency.

The security aspects of the explosion should be considered.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 14c

SSCP - Security and Safeguards Contingency Plan

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.3.1.2

Sub-Category: Fire / Explosion

Emergency Classification: **ALERT**

Replaced by 04-0 Rev 4 Scheme

Emergency Action Level:

Explosion affecting operability of one (1) train of safety systems.

Basis:

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

Only explosions that actually cause damage to equipment required for safe operation **AND** only damage that renders a single train of a safety system unable to perform its intended safety function meet the threshold of this EAL. A lengthy damage assessment should not be performed. The occurrence of the explosion with evidence of damage likely to prevent one train from performing its intended safety function is sufficient for declaration.

As used here, an explosion is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts significant energy to nearby structures and equipment.

If the explosion damages more than one train of a Safety System the event escalates to a Site Emergency.

The security aspects of the explosion should be considered.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 18c

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.3.1.3

Sub-Category: Fire / Explosion

Replaced By NEI Rev 4 Scheme

Emergency Classification: **Site Emergency**

Emergency Action Level:

Explosion affecting operability of two (2) trains of safety systems.

Basis:

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

Only explosions that actually cause damage to equipment required for safe operation of more than one safety system train **AND** only damage that affects the systems' ability to perform intended functions meet the threshold of this EAL. A lengthy damage assessment should not be performed. An immediate assessment of the probability of damage making multiple trains incapable of performing their safety function is all that is required. The occurrence of the explosion with evidence of damage likely to prevent the equipment in more than one train of a safety system from performing intended safety functions is sufficient for declaration.

As used here, an explosion is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts significant energy to nearby structures and equipment.

If only one train of a safety system is affected, see ALERT classification EAL.

The security aspects of the explosion should be considered.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 18c

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.3.2.1

Sub-Category: Fire / Explosion

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Fire within the Protected Area lasting more than 10 minutes after use of fire extinguishing equipment.

Basis:

The purpose of this EAL is to address fires which are potentially significant precursors to damage to safety systems. This condition applies to buildings or areas contiguous to plant vital areas or other significant buildings or areas.

Specifically excluded are small fires within administration buildings, wastebasket fires, or fires in areas of no safety consequence.

Escalation to a higher emergency class occurs if the fire affects one or more train(s) of a Safety System(s).

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 10

REPLACED BY NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.3.2.2

Sub-Category: Fire / Explosion

Emergency Classification: **ALERT**

Replaced By NEI 99-01 Rev 4 Scheme

Emergency Action Level:

Fire affecting operability of one (1) train of a safety system.

Basis:

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

This condition is entered when the Fire Brigade Leader reports a fire affects one train of a safety system or if Control Room Operators become aware of indications of impact to a safety system after a fire has been reported.

Only those fires that actually cause damage to equipment as reported by the Fire Brigade Leader or as noted by Control Room operators meet this EAL.

Escalation to a higher emergency class, if appropriate, is based on further system malfunctions, fission product barrier degradation, abnormal radiation levels, or Emergency Director judgment.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 13

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

REPLACED BY NEW EAL SCHEME

Category: Internal Events

EAL 5.3.2.3

Sub-Category: Fire / Explosion

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Fire affecting operability of two (2) trains of safety systems.

Basis:

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

This condition is entered when the Fire Brigade Leader reports a fire that affects more than one train of a safety system or if Control Room Operators become aware of indications of impact on more than one train of a safety system after a fire has been reported.

Only fires that actually cause damage to equipment required for safe operation of more than one safety system train **AND** only damage that affects the systems' ability to perform intended functions meet the threshold of this EAL. A lengthy damage assessment should not be performed. An immediate assessment of the probability of damage making multiple trains incapable of performing their safety function is all that is required. The occurrence of a fire with evidence of damage likely to prevent the equipment in more than one train of a safety system from performing intended safety functions is sufficient for declaration.

This condition is indicative of severe degradation of the level of safety at the plant with possible adverse consequences on the public health and safety. A Site Emergency is warranted.

Escalation to a higher emergency class, if appropriate, will be based on fission product barrier degradation or emergency management judgment.

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 11

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Internal Events

EAL 5.4.1.1

Sub-Category: Turbine Rotating Component Failures

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Visual confirmation of turbine housing penetration by a blade or rotating component.

Basis:

This initiation condition addresses the consequences of turbine failure and turbine missile effects.

Analyses documented in the FSAR on the consequences of turbine overspeed indicate that there would be only a low energy missile generated external to the low pressure turbine casing in the event of a turbine overspeed.

The study determined that the following components are subject to the possible effects of a turbine missile: one main steam line, the condensate storage tanks, reactor makeup water storage tanks, the reactor makeup water storage tank pumps, the refueling water storage tank, diesel generator fuel oil line, and the service water pump electrical leads. These components should be evaluated for damage.

Escalation to a higher emergency classification, if appropriate, is based on further missile damage from any source, system malfunctions, fission product barrier degradation, abnormal radiation levels, or emergency management judgment.

References:

WCAP 7525-L, Likelihood and Consequences of Turbine Overspeed at the Point Beach Nuclear Plant.

Reg Guide 1.115, Protection Against Low-Trajectory Turbine Missiles

FSAR 14.1.12, Likelihood of Turbine-Generator Unit Overspeed

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 14e and Alert 18e

REPLACED BY NEI 9901 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.1.1.1

Sub-Category: Natural Destructive Phenomena

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Any earthquake felt by Control Room Operators.	
OR	
An indicator light on two or more of the following Seismic Event Monitors	
SEI-6210	*#3 Warehouse*
SEI-6211	*Unit 1 Facade*
SEI-6212	*Drum Prep Room*
SEI-6213	*El. 8′ between vital switchgear room and aux feedwater tunnel*

Basis:

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 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 detectors of the plant are activated.

The seismic event monitors are set to alarm at 0.01g. Minor damage to some portions of the site may occur at these levels but should not affect the ability to safely operate the plant. Additional inspections may be desired to determine the extent of any damage. Therefore an Unusual Event classification is warranted.

This EAL requires two valid seismic alarms to eliminate classification due to plant operations or maintenance activities, such as heavy equipment moving near the monitor or an accidental impact to a monitor. Further validation may be accomplished by contacting the University of Wisconsin - Milwaukee Seismic Center.

Replaced by NEI 99-01 Rev Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

References:

PBNP FSAR, Appendix A

Setpoint Document STPT 22.1, Seismic Event Monitoring

EPRI Document, "Guidelines for Nuclear Plant Response to an Earthquake," dated October, 1989

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 13a

REPLACED BY NEI 99-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.1.1.2

Sub-Category: Natural Destructive Phenomena

Emergency Classification: **ALERT**

Emergency Action Level:

Valid Seismic Event Monitor readings of an intensity greater than 0.04g vertical or 0.06g horizontal.

Basis:

This EAL addresses events that may have resulted in the plant's vital equipment being subjected to forces beyond operational limits. Therefore an Alert classification is warranted. Classification should occur prior to a detailed damage assessment.

Values in this EAL are based on the Operating Basis Earthquake (OBE) limits (ground accelerations of .04g vertical and .06g horizontal) as defined by the FSAR.

Validation of seismic activity would be by severe ground shaking or by contacting University of Wisconsin - Milwaukee Seismic Center (Emergency Telephone Directory).

References:

PBNP FSAR, Appendix A

Setpoint Document STPT 22.1, Seismic Event Monitoring

EPRI Document, "Guidelines for Nuclear Plant Response to an Earthquake," dated October 1989

NUREG 0654, Appendix 1 Initiating Condition: Alert 17a

REPLACED NEI 99-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.1.1.3

Sub-Category: Natural Destructive Phenomena

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Valid Seismic Event Monitor readings of an intensity greater than 0.08g vertical or 0.12g horizontal.

Basis:

This EAL addresses events that may have resulted in the plant's vital equipment being subject to forces that may prevent safe shutdown and cooldown of the plant. Therefore a Site Emergency classification is warranted. Classification should occur prior to a detailed damage assessment.

Values in this EAL are based on the Safe Shutdown Earthquake (SSE) limits (ground accelerations of .08g vertical and .12g horizontal) as defined by the FSAR.

Validation of seismic activity would be by severe ground shaking or by contacting University of Wisconsin - Milwaukee Seismic Center (Emergency Telephone Directory).

References:

PBNP FSAR, Appendix A

Setpoint Document STPT 22.1, Seismic Event Monitoring

EPRI Document, "Guidelines for Nuclear Plant Response to an Earthquake," dated October 1989

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 15a

REPLACED BY NEI 99-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events **EAL 6.1.2.1**

Sub-Category: Natural Destructive Phenomena

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Any tornado visible from the site.

Basis:

This EAL is based on the assumption that a tornado may potentially damage plant or systems. An Unusual Event classification is warranted.

This condition is entered when a tornado is reported to the Shift Manager by plant personnel making visual observation.

Site is defined as the exclusion area which is the area within the site boundary surrounding PBNP in which the plant personnel have the authority to determine all activities including exclusion or removal of personnel and property from the area. At PBNP, the outer boundary of the exclusion area is coincident with site boundary. (Reference Appendix C of Emergency Plan).

If damage to safety-related equipment is confirmed (either by observation or plant instrumentation) the event may be escalated to an Alert. Other EALs should also be considered such as loss of electrical power.

References:

AOP-13C, Severe Weather Conditions

Probabilistic Safety Assessment -- High Winds, and Others Sec 9, Rev 0, Dated July 1995

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 13c

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Replaced By NEI 99-01 Rev 4 Scheme

Category: External Events

EAL 6.1.2.2

Sub-Category: Natural Destructive Phenomena

Emergency Classification: **ALERT**

Emergency Action Level:

Indications or observations that a tornado has damaged a vital structure. *OR* *Wind speed indicated as >90 MPH.*

Basis:

This EAL addresses events that may have resulted in a plant area being subjected to forces approaching or beyond design limits. It is assumed that damage may have occurred to plant safety systems. Therefore an Alert classification is warranted. Classification should occur prior to a detailed damage assessment.

The 100 MPH indicated wind speed was chosen as a value approaching the design basis for non-Class 1 metal structures at the plant. Although no damage to permanent plant structures should occur at this level, non-permanent structures (trailers, work shacks, temporary storage, etc.) could have significant damage and impact plant operations. Winds at this level would also impact personnel movement within and to the plant.

References:

AOP-13C, Severe Weather Conditions

FSAR 5.1, Containment System Structure

Probabilistic Safety Assessment -- High Winds, and Others Sec 9, Rev 0, Dated July 1995

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.

NUREG 0654, Appendix 1 Initiating Condition: Alert 17c

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events **EAL 6.1.2.3**

Sub-Category: Natural Destructive Phenomena

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Sustained winds greater than 100 MPH *AND* *Reports or indications of damage to vital equipment or structures.*

Basis:

This EAL addresses events that have resulted in plant areas being subjected to forces beyond design limits. It is assumed that substantial damage has occurred to plant structures with probable damage to safety systems.

It is inferred from Section 5.1 in the FSAR that the design straight wind speed of 108 mph was used in the design of the non-Class 1 metal structures. This is consistent with the Bechtel topical report. 100 mph was used in this EAL due to limitations of available instrumentation.

Therefore, this condition is indicative of serious plant system conditions with possible adverse consequences on the public health and safety. A Site Emergency is warranted.

Emergency classifications under other EALs may also be appropriate due to offsite effects caused by high winds, particularly status of offsite power lines.

Replaced by NP 9.0.1 Rev 4 Scheme

References:

AOP-13C, Severe Weather Conditions

FSAR 5.1, Containment System Structure

Probabilistic Safety Assessment -- High Winds, and Others Sec 9, Rev 0, Dated July 1995

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.

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 15c

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.2.1.1

Sub-Category: High Lake/Low Forebay Water Level

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Less than -11′ forebay or pump bay level with one unit's CW pumps off.

Basis:

This condition is considered a potential degradation in the level of safety of the plant due to Circulating Water Pumps and/or Service Water losing suction. Water levels at or below these levels impairs the ability of these pumps to provide water to their loads, and may result in subsequent loss of the safety function of the ultimate heat sink. Therefore, an Unusual Event classification is warranted.

References:

AOP-5A, Loss of Condenser Vacuum

AOP-13A, Circulating Water System Malfunction

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 13b

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.2.2.1

Sub-Category: High Lake/Low Forebay Water Level

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Any flooding which precludes access to the site or areas of the plant.

Basis:

This condition is considered to be a potential degradation in the level of safety of the plant due to limited access to the site or potential safety concerns for onsite personnel. Therefore an Unusual Event classification is warranted.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 13b

REPLACED BY NEI 99-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

REPLACED BY NEI (99-01) REV 4 SCHEME

Category: External Events

EAL 6.2.2.2

Sub-Category: High Lake/Low Forebay Water Level

Emergency Classification: **ALERT**

Emergency Action Level:

Flooding as indicated by greater than 6" of water in the 8 foot elevation of the Turbine Bldg.

Basis:

This EAL addresses an event that may result in a plant vital area being subjected to conditions beyond design limits adversely affecting plant safety systems. Therefore, this condition is indicative of abnormal plant conditions with possible adverse consequences on plant safety and is classified as an Alert.

This condition is entered when there is greater than six inches of water in the turbine hall. Although this EAL is in the category High Lake/Low Forebay, the cause of the flooding is not a factor. A broken Service Water or Circulating Water pipe could also create this condition.

The Turbine Building would flood before other plant areas, therefore it provides a representative indication of other possible problem areas. The feedwater pumps each sit on a base that is eight inches above the floor. The turbine seal oil pumps are approximately ten inches above the floor.

Escalation to a higher emergency class, if appropriate, will be based on Flooding in Vital Equipment Areas.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 17b

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events **EAL 6.2.2.3**

Sub-Category: High Lake/Low Forebay Water Level

Emergency Classification: **SITE EMERGENCY**

REPLACED BY NEI 99-01 SCHEME

Emergency Action Level:

Greater than 2′ of water in vital switchgear room *OR* *Greater than 2′ of water in auxiliary feedwater pump room.*

Basis:

This EAL addresses conditions where plant vital equipment may be subjected to conditions beyond design limits, and damage may be assumed to have occurred to plant safety systems. Therefore, this condition is indicative of serious plant system conditions with possible adverse consequences on the public health and safety. A Site Emergency is warranted.

Plant vital area designations are contained in the PBNP Security Plan.

Water levels in excess of two feet in the vital switchgear room severely threaten safe plant operations. Several 125-volt DC station batteries are installed in the vital switchgear room. The bottom and top of these batteries are 6 and 36 inches above the floor, respectively. Numerous electrical cabinets containing electrical components for the safety injection pumps, the station service transformers, and the 4.16 kV electrical system are also located in the room.

Water levels in excess of two feet in the auxiliary feedwater pump room threatens operation of the feedwater system and ultimately the ability to cool the reactor core. The turbine-operated auxiliary feedwater pumps are located approximately 18 inches above the floor and the motor operated auxiliary feedwater pumps are located approximately two feet above the floor. Additionally, the Source Range Output Expansion Control Panel is approximately two feet above the floor.

This EAL used to also contain criteria of greater than three feet of water in both EDG rooms, however this was before G03 and G04 were installed, hence spoke of G01 and G02 only. Due to the electrical arrangement of G03 and G04 as backups to G01 and G02 and the fact that G03 and G04 are at a significantly higher elevation, they have been removed from this EAL.

Emergency classifications under other EALs may be appropriate due to offsite effects caused by severe weather, particularly the status of offsite power lines.

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

References:

SOER 85-5, Internal Flooding of Power Plant Buildings

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 15b

REPLACED BY NEI 99-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.3.1.1

Sub-Category: Toxic/Flammable Gas Intrusion

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Near or on-site flammable or toxic gas release as reported to Shift Manager by plant personnel making visual observation.

Basis:

The release of toxic or flammable gas in or near the Exclusion Area may pose a potential threat to reactor plant and personnel safety. It is the potential threat to normal operation or hazard to personnel which must be evaluated. If no such threat exists, the EAL is not met. If, however, personnel safety or plant operation is threatened, an Unusual Event is warranted.

Flammable gases are typically more limiting than toxic gases. Although an SCBA could protect from toxicity, detonation of a flammable gas could be immediately hazardous to personnel.

On-site is defined as the exclusion area which is the area within the site boundary surrounding PBNP in which the plant personnel have the authority to determine all activities including exclusion or removal of personnel and property from the area. At PBNP, the outer boundary of the exclusion area is coincident with the site boundary.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 14d

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.3.1.2

Sub-Category: Toxic/Flammable Gas Intrusion

Emergency Classification: **ALERT**

REPLACED BY NEW 01 Rev 1 SCHEME

Emergency Action Level:

Entry of toxic or flammable gas into a plant building atmosphere affecting operation or access.

Basis:

The release of toxic or flammable gas significant enough to affect plant operation (i.e., initiate a plant transient or preclude access to plant equipment) warrants declaration of an Alert:

Flammable gases are typically more limiting than toxic gases. Although an SCBA could protect from toxicity, detonation of a flammable gas could be immediately hazardous to personnel. An area where access is not required for plant operation, which could be evacuated, does not warrant an Alert, but may warrant an Unusual Event if the potential exists to affect operation or personnel.

Any affected area normally accessed for plant operation (PAB, Turbine hall, etc.) meets the Alert level. If vital areas are affected, see EAL 6.3.1.3.

The primary flammable gases considered are acetylene, propane and hydrogen.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 18d

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.3.1.3

Sub-Category: Toxic/Flammable Gas Intrusion

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Entry of toxic or flammable gas into a plant vital area affecting operation or personnel safety *AND* *Reactor coolant temperature greater than 200°F.*

Basis:

The release of toxic or flammable gas into a plant vital area poses a significant threat to plant safety by precluding access to plant vital equipment which may be needed for Safe Shutdown. Therefore this condition warrants declaration of a Site Emergency.

Flammable gases are typically more limiting than toxic gases. Although an SCBA could protect from toxicity, detonation of a flammable gas could be immediately hazardous to personnel.

This EAL does not apply in cold shutdown or refueling modes due to the significantly reduced probability that the loss of access would result in fuel failure and/or a release.

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 16c

REPLACED BY NEI 99-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.4.1.1

Sub-Category: Vehicle/Missile Impacts

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Unusual aircraft activity over facility.

Basis:

This event may warrant the prompt notification of state and local authorities and perhaps a precautionary notification of Emergency Response Organization personnel. This event could pose a potential threat to plant operation or personnel safety and therefore warrants declaration of an Unusual Event.

The Protected Area Physical Barrier is defined in the PBNP Security Plan. Note: The Independent Spent Fuel Storage Installation (ISFSI) is a separate Protected Area.

References:

NUREG 0654, Appendix 1 Initiating Condition, Unusual Event 14a

SSCP - Security and Safeguards Contingency Plan

Replaced By NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.4.1.2

Sub-Category: Vehicle/Missile Impacts

Emergency Classification: **ALERT**

Emergency Action Level:

Aircraft crash in Protected Area (within the fence)

Basis:

This condition is entered when Control Room Operators become aware of an aircraft crash in the Protected Area (within the fence).

A lengthily damage assessment should not be performed. The occurrence of a crash is sufficient for declaration.

The ISFSI is part of the protected area.

This condition is indicative of abnormal plant system conditions with possible adverse consequences on the public health and safety is classified as an ALERT.

Escalation to a higher emergency class, if appropriate, will be based on further system malfunctions, fission product barrier degradation, abnormal radiation levels, or Emergency Director judgment.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 18a

Replaced By NEI 99-01 Rev 4 scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.4.1.3

Sub-Category: Vehicle/Missile Impacts

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Aircraft crash affecting operability of two (2) trains of safety systems.

Basis:

This condition is indicative of severe degradation of the level of safety at the plant and with possible adverse consequences on the public health and safety is classified as a Site Emergency.

Only crashes that actually cause damage to equipment required for safe operation of more than one safety system train **AND** only damage that affects the systems' ability to perform intended functions meet the threshold of this EAL. A lengthy damage assessment should not be performed. The occurrence of a crash with evidence of damage likely to prevent the equipment in more than one train of a safety system from performing intended safety functions is sufficient for declaration.

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 16a

Replaced by EAL-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.4.2.2

Sub-Category: Vehicle/Missile Impacts

Emergency Classification: **ALERT**

Emergency Action Level:

Missile impact from any source by visual observation of Operations Supervisor.

Basis:

This condition is entered when Operations Supervision becomes aware of a missile impact.

A lengthily damage assessment should not be performed. The occurrence of a missile impact is sufficient for declaration.

This condition is indicative of abnormal plant system conditions with possible adverse consequences on the public health and safety is classified as an ALERT.

Escalation to a higher emergency class, if appropriate, will be based on further system malfunctions, fission product barrier degradation, abnormal radiation levels, or emergency management judgment.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 18b

Replaced By NEI 99-01 Rev 4 scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: External Events

EAL 6.4.2.3

Sub-Category: Vehicle/Missile Impacts

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Any missile impact affecting operability of two (2) trains of safety systems.

Basis:

Safety systems as used here designates systems with safety-related functions. Attachment D lists safety systems and systems with safety-related functions.

Only missile impacts that actually cause damage to equipment required for safe operation of more than one safety system train **AND** only damage that affects the systems' ability to perform intended functions meet the threshold of this EAL. A lengthy damage assessment should not be performed. An immediate assessment of the probability of damage making multiple trains incapable of performing their safety function is all that is required. The occurrence of a missile impact with evidence of damage likely to prevent the equipment in more than one train of a safety system from performing intended safety functions is sufficient for declaration.

Major losses of plant safety systems, as defined by failure of the ability of two or more of the safety systems to perform their intended function, warrants declaration of a Site Emergency.

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 16b

REPLACED BY NA1-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fuel Handling/ISFSI Events

EAL 7.1.1.2

Sub-Category: Fuel Handling Events

Emergency Classification: **ALERT**

Emergency Action Level:

Report of possible damage to irradiated fuel combined with an alarm on any of the following radiation monitors

RE-211, Containment air particulate monitor

RE-212 Containment noble gas monitor

RE-221 Drumming Area Vent

Manipulator Area Monitor

Spent Fuel Bridge Area Monitor.

Replaced By NEI 99-01 Rev 4 Scheme

Basis:

A report of possible damage to irradiated fuel, combined with an alarm on any of the radiation monitors indicates the probable damage to spent fuel.

NUREG/CR-4982 states that even if no corrective actions are taken, no prompt fatalities are predicted and the risk of injury is low. In addition, NRC Information Notice No. 90-08 presents the following clarifications:

"In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel."

An Alert classification is appropriate for this event. Escalation would be based on actual radiological releases and/or SM judgment.

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

References:

NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82"

NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel"

AOP-8B, Irradiated Fuel Handling Accident in Containment

AOP-8C, Fuel Handling Accident in Primary Auxiliary Building

NUREG 0654, Appendix 1 Initiating Condition: Alert 12

Replaced By NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fuel Handling/ISFSI Events

EAL 7.2.1.2

Sub-Category: Irradiated Fuel Events

Emergency Classification: **ALERT**

REPLACED BY NEI 99-01 REV 4 SCHEME

Emergency Action Level:

Indications of irradiated fuel uncovered.

Basis:

This EAL applies to any area where irradiated fuel is located; reactor cavity, reactor vessel, or the spent fuel pool.

Any releases caused by uncovering the fuel are not generally the primary concern. The primary concern of this EAL is two-fold. First, is the evident loss of control of inventory. The second is the immediate, life threatening dose which could be present in the area due to loss of shielding.

An Alert classification is appropriate for this event. Escalation, if required, would be based on actual radiological releases or Emergency Director judgment.

This EAL applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

References:

NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82"

NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel"

AOP-8F, Loss of Spent Fuel Pool Cooling

NUREG 0654, Appendix 1 Initiating Condition: Alert 12

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fuel Handling/ISFSI Events

EAL 7.3.1.1

Sub-Category: ISFSI Events

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

A loaded spent fuel cask dropped or tipped.

Basis:

The Independent Spent Fuel Storage Installation (ISFSI) stores spent fuel in vertical casks outside the main Protected Area. Engineering safeguards and procedures insure these casks are not dropped or tipped for the duration of their expected lifetimes. If they should be dropped or tipped it is appropriate to declare an Unusual Event until the situation is analyzed and corrected.

References:

NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82"

NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel"

AOP-8G, Ventilated Storage Cask (VSC) Drop or Tipover

NUREG 0654, Appendix 1 Initiating Condition: Alert 12

Replaced By NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Fuel Handling/ISFSI Events

EAL 7.3.1.2

Sub-Category: ISFSI Events

Emergency Classification: **ALERT**

Emergency Action Level:

Breach of a loaded spent fuel cask as indicated by a reading of greater than 1000 mRem/hr at 1 meter.

Basis:

The Independent Spent Fuel Storage Installation (ISFSI) stores spent fuel bundles in vertical cask in an area outside the main Protected Area. Engineering safeguards and procedures are in place to insure these casks are not subjected to forces that could breach their integrity. If a cask is breached it is appropriate to declare an Alert due to the potential threat to site personnel.

References:

NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82"

NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel"

NUREG 0654, Appendix 1 Initiating Condition: Alert 12

Replaced by NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Emergency Management Judgment **EAL 8.1.1.1**

Sub-Category: None

Emergency Classification: **UNUSUAL EVENT**

Emergency Action Level:

Any event which in the judgment of the Shift Manager or the Emergency Director could lead to, or has led to, a potential degradation of the level of safety of the plant.

Basis:

This EAL would pertain to conditions not explicitly addressed elsewhere in the EALs, but which warrant the declaration of an emergency due to the potential degradation of the level of safety of the plant. The Shift Manager or Emergency Director makes this determination.

References:

NUREG 0654, Appendix 1 Initiating Condition: Unusual Event 15

REPLACED BY NEI 99-01 Rev 4 SCHEME

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Emergency Management Judgment

EAL 8.1.1.2

Sub-Category: None

Emergency Classification: **ALERT**

Replaced by NEI 99-01 Rev 4 Scheme

Emergency Action Level:

Any event which in the judgment of the Shift Manager or the Emergency Director could cause or has caused actual or potential substantial degradation of the level of safety of the plant.

Basis:

This EAL would pertain to conditions not explicitly addressed elsewhere in the EALs, but which warrant the declaration of an emergency due to the actual or substantial potential degradation of the level of safety of the plant. The Shift Manager or Emergency Director makes this determination.

In keeping with other EALs, generally events which challenge single (RCS or Fuel Cladding) barriers, or affect only single safety systems or functions fall in this category.

References:

NUREG 0654, Appendix 1 Initiating Condition: Alert 19

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Emergency Management Judgment **EAL 8.1.1.3**

Sub-Category: None

Emergency Classification: **SITE EMERGENCY**

Emergency Action Level:

Any event which in the judgment of the Shift Manager or the Emergency Director could indicate actual or likely major failures of plant functions needed to protect the public. Any releases are not expected to result in exposures in excess of EPA PAGs.

Basis:

This EAL would pertain to conditions not explicitly addressed elsewhere in the EALs, but which warrant the declaration of an emergency due to the actual or likely failure of major plant functions needed for the protection of the public. The Shift Manager or Emergency Director makes this determination.

In keeping with other EALs, generally events which challenge two barriers (but not three), or affect more than one safety system or safety function, fall into this category.

References:

NUREG 0654, Appendix 1 Initiating Condition: Site Emergency 17

Replaced By NEI 99-01 Rev 4 Scheme

ATTACHMENT B
EMERGENCY ACTION LEVELS (EALs)

Category: Emergency Management Judgment **EAL 8.1.1.4**

Sub-Category: None

Emergency Classification: **GENERAL EMERGENCY**

Emergency Action Level:

Any event which in the judgment of the Shift Manager or the Emergency Director could lead to actual or imminent core damage and the potential for a large release of radioactive material (in excess of EPA PAGs) outside the site boundary.

Basis:

This EAL pertains to conditions not explicitly addressed elsewhere in the EALs but which warrant declaration of an emergency due to actual or imminent core damage and the potential exists for a release of large amounts of radioactive material. The Shift Manager or Emergency Director makes this determination.

In keeping with other EALs, generally events which challenge all three barriers, indicate the potential for core damage, or which reflect possible large releases fall into this category.

References:

NUREG 0654, Appendix 1 Initiating Condition: General Emergency 4 and 7

REPLACED BY NEI 99-01 Rev SCHEME

ATTACHMENT C
FISSION PRODUCT BARRIER (FPB) MATRIX

This attachment is used to determine the status of the three primary Fission Product Barriers as they relate to classification. Wherever possible existing well-known parameters have been selected as thresholds for determining the status of the barriers. This is to integrate setpoints and thresholds already in existence in EOPs and Critical Safety Status Trees into the classification process. The intended purpose is to minimize the number of separate limits and values.

NOTE: ***Do not "anticipate" challenge or loss of a barrier unless the trend is rapid, and the values are close to the threshold/criteria.***

The table on the following page may be used to 'check off' the status of the three Fission Product Barriers. Next to each code (FC-1, RL-2, etc.) is an empty box. If the plant conditions meet the conditions in the box, the associated box may be checked, either in the Challenged or Loss column.

The number and status of Fission Product Barriers may then be compared to the EALs that specifically address Fission Product Barrier status (Category 1 of Attachment A).

- Generally, one barrier LOST is an Alert (unless the barrier is Containment alone),
- two barriers LOST is a Site Emergency and
- two barriers LOST, with a CHALLENGE or LOSS of the third barrier is a General Emergency.

The codes (FC-1, RL-2, etc.) may be used to obtain further explanation as to the basis of their development. Each initial code letter; 'F' for Fuel Cladding, 'R' for Reactor Coolant System, or 'C' for Containment is followed by either 'C' for Challenge or 'L' for Loss. (For example FL-# indicates a parameter for Fuel Cladding LOSS, RC-# indicates a parameter for Reactor Coolant System Challenge.) The bases are on the pages following the Table, arranged by barrier, Challenge then Loss.

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT C
FISSION PRODUCT BARRIER (FPB) MATRIX

Replaced by [illegible] 04-01 Rev 4 [illegible]

		FUEL CLAD CHALLENGE			FUEL CLAD LOSS
☐	FC-1	ST-2 (Core Cooling) Orange Path. **Degraded** core cooling as indicated by ANY of the following: • CET <700°F AND reactor <25′ NR • CET >700°F AND reactor >25′ NR • Reactor vessel <[120]110′ WR with 2 RCPs OR <[60]50′ with 1 RCP.	☐	FL-1	ST-2 (Core Cooling) Red Path. **Inadequate** core cooling as indicated by EITHER: • CETs >1200°F Also SAMG entry. • CETs > 700°F and reactor vessel level <25′ NR
☐	FC-2	Failed fuel monitor (RE-109) reading greater than 120 mRem/hr.	☐	FL-2	Failed fuel monitor (RE-109) reading greater than 600 mRem/hr.
☐	FC-3	2 of 3 containment high range monitors reading greater than 1000 Rem/hr.	☐	FL-3	2 of 3 containment high range monitors reading greater than 6000 Rem/hr.
☐	FC-4	Coolant activity greater than Technical Specification TS 3.4.16 (TSAC 3.4.16.B or 3.4.16.C entered).	☐	FL-4	Coolant activity greater than 250 µCi/gram equivalent of I-131
☐	FC-5	Any condition which in the judgment of the Emergency Director is indicative of a challenge to the Fuel Cladding barrier.	☐	FL-5	Any condition which in the judgment of the Emergency Director is indicative of a loss of the Fuel Cladding barrier.
		RCS CHALLENGE			**RCS LOSS**
☐	RC-1	RCS leak greater than 10 gpm or 500 gallons per day in either steam generator (Technical Specifications).	☐	RL-1	RCS leak greater than 50 gpm. If greater than 400 gpm, see also EAL 2.4.1.3
☐	RC-2	ST-4 (Integrity) Orange Path. Excess RCS cooldown or cold overpressurization of the RCS.	☐	RL-2	ST-4 (Integrity) Red Path. Temperature in either cold leg <285°F and cooldown >100°F in the last 60 minutes.
☐	RC-3	Any condition which in the judgment of the Emergency Director is indicative of a challenge to the Reactor Coolant System barrier.	☐	RL-3	Any condition which in the judgment of the Emergency Director is indicative of a loss of the Reactor Coolant System barrier.
		CONTAINMENT CHALLENGE			**CONTAINMENT LOSS**
☐	CC-1	ST-5 (Containment) Orange Path. Containment pressure >25 psig and increasing following actuation of containment spray OR Sump 'B' >74"	☐	CL-1	ST-5 (Containment) Red Path. Pressure >60 psig
☐	CC-2	Hydrogen concentration greater than 2%.	☐	CL-2	Hydrogen concentration greater than 4%.
☐	CC-3	Atmospheric dump(s) or reliefs open and greater than 10 gpm Primary to Secondary leakage exists	☐	CL-3	Unisolable steam line break outside containment. If primary to secondary leakage >10 gpm exists, see also RC-1. If >50 gpm, see RL-1.
☐	CC-4	Any condition which in the judgment of the Emergency Director is indicative of a challenge to Containment barrier.	☐	CL-4	Inability to isolate Containment.
			☐	CL-5	Any condition which in the judgment of the Emergency Director is indicative of a loss of the Containment barrier.

REFERENCE USE

ATTACHMENT C
FISSION PRODUCT BARRIER (FPB) MATRIX

Fuel Cladding - CHALLENGE

FC-1

ST-2 (Core Cooling) Orange Path indicates that RCS subcooling has been lost as well as loss of RCS inventory. RCS subcooling and reactor vessel level are fundamental indications of the assurance of adequate core cooling. These conditions indicate a challenge to the fuel cladding barrier due to **degraded** core cooling.

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

FC-2

The function of the failed fuel monitor is to monitor coolant activity. As the fuel cladding barrier degrades increasing amounts of activity are present in the coolant, and seen by this monitor. The value selected is approximately Technical Specifications, hence an Unusual Event must also be declared at this value, if this is the only fission product barrier affected. (1.1.2.1)

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

FC-3

In-containment high radiation monitors monitor activity in the coolant. As the fuel cladding barrier degrades increasing amounts of activity are present in the coolant, and seen by these monitors. This parameter may be the first indication of cladding degradation due to the location of the failed fuel monitor and possible containment isolation. The value is not correlated to a specific percentage of clad damage.

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

FC-4

Coolant activity greater than Technical Specifications is considered a precursor to loss of the fuel cladding barrier. (TSAC 3.4.16.B or 3.4.16.C has been entered).

For purposes of emergency classification, the barrier is to be considered CHALLENGED.

FC-5

It is unlikely that any classification scheme can anticipate every circumstance. Therefore this 'threshold' criteria is based on an ad hoc judgment call. If the Emergency Director has reason to believe the integrity of this barrier is being challenged, he may declare it so.

He should have objective reason to believe the barrier is challenged. Simply not knowing (for example loss of indications) should <u>not</u> be used as a basis for declaring a barrier challenged or lost.

If the barrier is subsequently determined not to have been challenged, it may be declared intact.

If it is determined that the barrier was challenged, but is no longer challenged, the barrier must remain as challenged, until the Recovery phase of the emergency.

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

ATTACHMENT C
FISSION PRODUCT BARRIER (FPB) MATRIX

Fuel Cladding – LOSS

FL-1
ST-2 (Core Cooling) Red Path indicates that RCS subcooling has been lost as well as significant loss of RCS inventory. RCS subcooling and reactor vessel level are fundamental indications of the assurance of adequate core cooling. These conditions indicate the fuel cladding barrier has been subjected to conditions which may cause its failure due to **inadequate** core cooling.. For the purposes of emergency classification, the barrier is to be considered LOST. Core exit thermocouple reading in excess of 1200°F is also an entry condition for Severe Accident Management Guidelines (SAMGs).

FL-2
The function of the failed fuel monitor is to monitor coolant activity. As the fuel cladding barrier degrades increasing amounts of activity are present in the coolant, and seen by this monitor. The value is not correlated to a specific percentage of clad damage, but is beyond Technical Specifications. For the purposes of emergency classification, the barrier is to be considered LOST.

FL-3
In-containment high radiation monitors monitor activity in the coolant. As the fuel cladding barrier degrades increasing amounts of activity are present in the coolant, and seen by these monitors. This parameter may be the first indication of cladding degradation due to the location of the failed fuel monitor and possible containment isolation. The value is not correlated to a specific percentage of clad damage, but is beyond Technical Specifications. For the purposes of emergency classification, the barrier is to be considered LOST.

FL-4
Coolant activity greater than this level is not correlated to a specific percentage of clad damage, but is beyond Technical Specifications. For the purposes of emergency classification, the barrier is to be considered LOST.

FL-5
It is unlikely that any classification scheme can anticipate every circumstance. Therefore this 'threshold' criteria is based on an ad hoc judgment call. If the Emergency Director has reason to believe the integrity of this barrier is lost, he may declare it so. He should have objective reason to believe the barrier is lost. Simply not knowing (for example loss of indications) should <u>not</u> be used as a basis for declaring a barrier challenged or lost. If the barrier is subsequently determined not to have been lost, it may be declared intact, or challenged, as appropriate. If it is determined that the barrier was lost, but is no longer lost, the barrier must remain as lost, until the Recovery phase of the emergency. For the purposes of emergency classification, the barrier is to be considered LOST.

Replaced By NEI Rev 4 Scheme

ATTACHMENT C
FISSION PRODUCT BARRIER (FPB) MATRIX

Reactor Coolant System - CHALLENGE

RC-1

These conditions represent minor leakage from the RCS. Because the source of the leak may not be known, and leaks can become worse, these conditions are considered precursors to more serious events. As such, an Unusual Event must be declared on these conditions, if the RCS is the only barrier affected. (1.1.3.1, 1.1.4.1)

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

RC-2

Conditions of ST-4 (Integrity) Orange Path reflect an excessive cooldown of the vessel or cold overpressurization of the RCS. These conditions represent a challenge to the RCS barrier. An Unusual Event must be declared. (1.1.5.1)

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

RC-3

It is unlikely that any classification scheme can anticipate every circumstance. Therefore this 'threshold' criteria is based on an ad hoc judgment call. If the Emergency Director has reason to believe the integrity of this barrier is being challenged, he may declare it so.

He should have objective reason to believe the barrier is challenged. Simply not knowing (for example loss of indications) should not be used as a basis for declaring a barrier challenged or lost.

If the barrier is subsequently determined not to have been challenged, it may be declared intact.

If it is determined that the barrier was challenged, but is no longer challenged, the barrier must remain as challenged, until the Recovery phase of the emergency.

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT C
FISSION PRODUCT BARRIER (FPB) MATRIX

Reactor Coolant System – LOSS

REPLACED BY NEI-99-01 REV 4 SCHEME

RL-1

This value is derived from NUREG-0654, Appendix 1. Although 50 gpm is well within the capacity of available pumps, this leak can be either into Containment or from Primary to Secondary systems. Thus, the RCS barrier is no longer serving its function of preventing the transport of fission products.

For the purposes of emergency classification, the barrier is to be considered LOST.

RL-2

Conditions of ST-4 (Integrity) Red Path reflect an excessive cooldown of the vessel. These conditions indicate the RCS barrier has been subjected to conditions which may cause its failure

For the purposes of emergency classification, the barrier is to be considered LOST.

RL-3

It is unlikely that any classification scheme can anticipate every circumstance. Therefore this 'threshold' criteria is based on an ad hoc judgment call. If the Emergency Director has reason to believe the integrity of this barrier is lost, he may declare it so.

He should have objective reason to believe the barrier is lost. Simply not knowing (for example loss of indications) should not be used as a basis for declaring a barrier challenged or lost.

If the barrier is subsequently determined not to have been lost, it may be declared intact, or challenged, as appropriate.

If it is determined that the barrier was lost, but is no longer lost, the barrier must remain as lost, until the Recovery phase of the emergency.

For the purposes of emergency classification, the barrier is to be considered LOST.

ATTACHMENT C
FISSION PRODUCT BARRIER (FPB) MATRIX

Containment - CHALLENGE

CC-1

ST-5 (Containment) Orange Path represent conditions beyond normal operating parameters due to either pressure or sump "B" level.

For the purposes of emergency classification, the barrier is to be considered CHALLENGED

CC-2

Existence of hydrogen at these concentrations does not yet represent an explosive mixture, however, there are limited means to reduce hydrogen in containment, especially during an emergency.

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

CC-3

This challenge threshold is designed to ensure that if Fuel Cladding **AND** RCS barriers are LOST, a General Emergency would be declared if the atmospheric dump valves or relief valves on the affected steam generator open (or are opened) and greater than 10 gpm Primary to Secondary leakage exists. If the Primary to Secondary leakage is less than 10 gpm the RCS barrier may be considered intact.

This threshold is included to address NUREG-0654, Appendix 1 Initiating Condition A4.

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

CC-4

It is unlikely that any classification scheme can anticipate every circumstance. Therefore this 'threshold' criteria is based on an ad hoc judgment call. If the Emergency Director has reason to believe the integrity of this barrier is being challenged, he may declare it so.

He should have objective reason to believe the barrier is challenged. Simply not knowing (for example loss of indications) should not be used as a basis for declaring a barrier challenged or lost.

If the barrier is subsequently determined not to have been challenged, it may be declared intact.

If it is determined that the barrier was challenged, but is no longer challenged, the barrier must remain as challenged, until the Recovery phase of the emergency.

For the purposes of emergency classification, the barrier is to be considered CHALLENGED.

Replaced By NEI 99-01 Rev 4 Scheme

ATTACHMENT C
FISSION PRODUCT BARRIER (FPB) MATRIX

REPLACED BY NEW 01-1 REV 4 SCHEME

Containment - LOSS

CL-1
ST-5 (Containment) Red Path represent conditions indicate the containment barrier has been subjected to conditions which may cause its failure. For the purposes of emergency classification, the barrier is to be considered LOST.
CL-2
Hydrogen at these concentrations may detonate. This would create an explosion in Containment. For the purposes of emergency classification, the barrier is to be considered LOST.
CL-3
Main steam line piping outside containment, up to and including the isolation valves may be considered a part of the Containment barrier. The inability to isolate assumes it is desired and has been attempted. This attempt includes only actions which may be taken from the Control Room. If actions must be taken outside the Control Room to isolate, the barrier must be considered lost. For the purposes of emergency classification, the barrier is to be considered LOST.
CL-4
This criteria includes all isolation paths, including access hatches. Only one valve or door in a given path need be closed. A physical loss of integrity (crack or hole) also meets this criteria. For the purposes of emergency classification, the barrier is to be considered LOST.
CL-5
It is unlikely that any classification scheme can anticipate every circumstance. Therefore this 'threshold' criteria is based on an ad hoc judgment call. If the Emergency Director has reason to believe the integrity of this barrier is lost, he may declare it so. He should have objective reason to believe the barrier is lost. Simply not knowing (for example loss of indications) should not be used as a basis for declaring a barrier challenged or lost. If the barrier is subsequently determined not to have been lost, it may be declared intact, or challenged, as appropriate. If it is determined that the barrier was lost, but is no longer lost, the barrier must remain as lost, until the Recovery phase of the emergency. For the purposes of emergency classification, the barrier is to be considered LOST.

ATTACHMENT D
SAFETY AND SAFETY-RELATED SYSTEMS

Designator	System	Safety-Related Functions
AF	Auxiliary Feedwater	Feedwater supply
AMSAC	ATWS Mitigation System Actuation Circuitry	Non-safety-related isolation
BS	Boron Recycle	Supports Safety Injection
CC	Component Cooling Water	Cools safety-related equipment
CI	Containment Integrity	Containment integrity
COMP	Computers	Monitoring
CONT	Containment Structures	Containment integrity
CP	Containment Penetrations	Containment integrity
CS	Condensate and Feedwater	Core Cooling
CV	Chemical and Volume Control	Emergency cooling, containment integrity
DA	Diesel Starting Air	Diesel start
DG	Diesel Generator	Emergency power supply
ESF	Engineered Safety Features (Safeguards)	Core Cooling and Integrity
FH	Fuel Handling	Fuel integrity
FM	In core Flux Mapping	Fuel integrity
FO	Fuel Oil	Power supply
FP	Fire Protection	Fire Protection
FW	Feedwater (I&C only)	Non-safety-related isolation
HV	Auxiliary Steam, Heating Steam & Condensate, Chilled & Hot Water	Containment cooling and integrity

REPLACED BY NEI 99-01 Rev [illegible] Scheme

ATTACHMENT D
SAFETY AND SAFETY-RELATED SYSTEMS

Designator	System	Safety-Related Functions
IA	Instrument Air	Containment isolation and integrity
IST	Inservice Test Equipment (i.e., steam generator nozzle dams)	Reactor coolant system integrity
MRR	Metering, Relaying, & Regulation	Monitoring
MS	Main, Extraction, Gland Seal & Reheat Steam	Containment integrity, heat removal
NG	Nitrogen Gas	Monitoring
NI	Nuclear Instrumentation	Reactor protection
PACV	Post-Accident Vent, Drains, etc.	Containment integrity, containment hydrogen control
PPCS	Plant Process Computer System	Monitoring
RC	Reactor Coolant	Reactor coolant system integrity, reactor protection, containment integrity
RDC	Rod Drive Control	Reactor coolant system integrity, reactor protection
RH	Residual Heat Removal (LPSI)	Containment integrity, emergency cooling
RM	Radiation Monitoring	Monitoring RCS and containment integrity
RP	Reactor Protection	Reactor protection, monitoring,
RS	Radwaste Steam	Non-safety-related isolation
S	Structures	Safety-related equipment safety
SA	Service Air	Containment integrity
SF	Spent Fuel Cooling and Filtration	Heat removal and containment integrity
SC	Primary Sampling	Containment and RCS integrity
SI	Safety Injection (HPSI)	Emergency cooling, heat removal, containment integrity

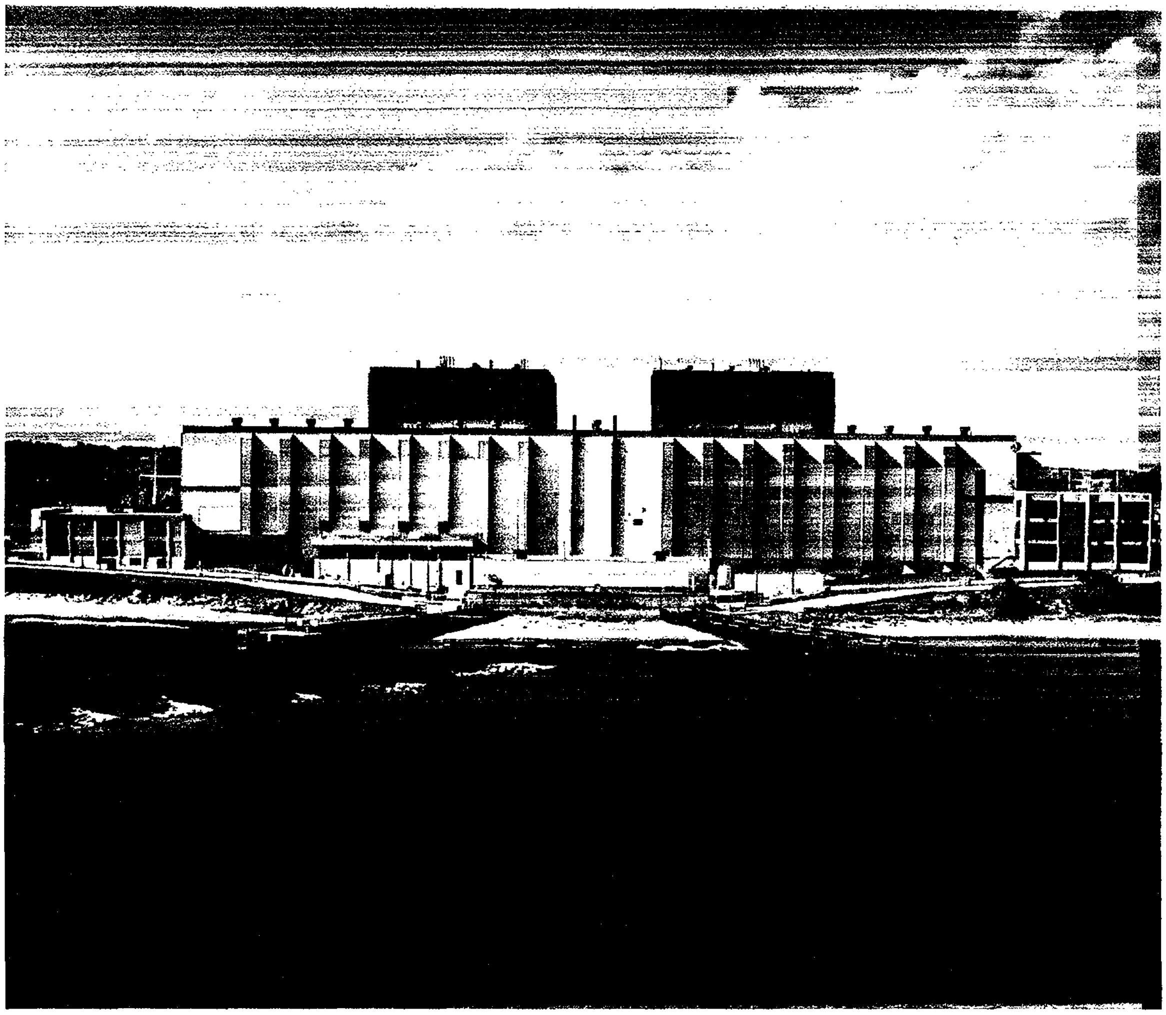
REPLACED BY NEI 99-01 REV 4 SCHEME

ATTACHMENT D
SAFETY AND SAFETY-RELATED SYSTEMS

Designator	System	Safety-Related Functions
SW	Service Water	Feedwater supply, heat removal, containment integrity
VNBI	PAB Battery & Inverter Room H&V	Heat removal, battery room hydrogen control
VNCC	Containment Accident Fans H&V	Heat removal
VNDG	Diesel Generator Room H&V	Support Diesel operation
VNPSE	Containment Purge Supply & Exhaust H&V	Containment integrity
VNRC	Reactor Cavity Cooling H&V	Containment integrity
WG	Waste Gas	Containment integrity
WL	Waste Liquid	Containment integrity
Y	Vital Instrument Bus 120 VAC	Power supply
4.16KV	4160V Electrical	Power supply
480V	480V Electrical	Power supply
125V	125VDC Electrical	Power supply

REPLACED BY NEW EPIP-01 REV [illegible] ENTIRETY

ENCLOSURE II



PROPOSED POINT BEACH EAL'S

NEI 99-01 REV.4
SCHEME

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: ____

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 1.13 CA052693 MG 4/20/04

Document Preparer (print/sign) Pat Schwartz / Pat Schwartz Date 04/15/2004

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) Todd M. Gemskie / Todd M Gemskie Date 5/3/04

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 / Monica Ray Date 5/3/04

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) Jim Shaw / [illegible] Date 5/12/04 5/19/04 [illegible] 6/15/04

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. 2004-036/2004-039

Approval Authority (print/sign) MONICA RAY / Monica Ray Date 6/16/04

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): ____

Point Beach Nuclear Plant

DOCUMENT REVIEW AND APPROVAL CONTINUATION

Doc Number **Emergency Plan Appendix B** Revision **22** Unit **PBO**

Title **Emergency Classification**

Temporary Change Number ________

Description of Changes:

Step *	**Change/Reason**
Total Rewrite	Replaced Emergency Classification Chart with 2 new Emergency Classification Charts – 1 for Hot Conditions (RCS greater than 200 degrees Fahrenheit) and 1 for Cold Conditions (RCS less than or equal to 200 degrees Fahrenheit.

Other Comments

* Note: Recording of Step Number(s) is not required for multiple occurrences of identical information or when not beneficial to reviewers.

10 CFR 50.59/72.48 APPLICABILITY FORM

Brief Activity Title or Description: **Revise EP Appendix B**

This form is required to be completed and attached to the applicable activity change forms to document all or portions of an activity that are covered by another regulation other than 10 CFR 50.59 and 10 CFR 72.48 (pre-screening criteria 2). See NP 5.1.8, 10 CFR 50.59/72.48 Applicability, Screening and Evaluation (New Rule).

NOTE: **Guidance for searching the FSAR, Technical Specifications, Regulatory Commitments (CLB Commitment Database) and other licensing basis documents can be found in NP 5.1.8, Attachment G.**

NOTE: **Although 10 CFR 50.59 and 72.48 may not be applicable to the processes listed below, change activities conducted under these processes may require changes to the FSAR. If so, initiate FSAR changes per NP 5.2.6, FSAR Revisions.**

	Regulatory or Plant Process	YES	NO
1.	Does the activity require a change to the Facility Operating License, License Conditions or Technical Specifications? (If the answer is <u>YES</u>, process the applicable changes per NP 5.2.7, License Amendment Request Preparation, Review and Approval.)	☐	☒
2.	**NOTE: The Quality Assurance Plan is described in FSAR Section 1.4.** Does the activity require a change to the Quality Assurance Program? If the answer is <u>YES</u>, process the applicable changes per NP 11.1.3, QA Program Revisions.	☐	☒
3.	**NOTE: Implementation of Security Plan changes that require physical changes to the plant, or changes to operator access to the plant require a screening.** **NOTE: Security is described in FSAR Section 12.7.** Does the activity require a change to the PBNP Security Plan, a safeguards contingency plan, or security training and qualification plan? If the answer is <u>YES</u>, assess the acceptability of the change per 10 CFR 50.54(p) using Security procedures.	☐	☒
4.	**NOTE: The Emergency Plan is described in FSAR Section 12.6.** Does the activity require a change to the Emergency Plan? If the answer is <u>YES</u>, assess the acceptability of the change per 10 CFR 50.54(q) using NP 1.8.3, 10 CFR 50.54(q) Evaluations.	☒	☐
5.	**NOTE: The Radiation Protection Program is described in FSAR Section 11.4.** Does the activity involve a change to the PBNP Radiation Protection Program or its implementing procedures, <u>AND</u> the activity in its entirety is within the requirements of 10 CFR 20, Standards for Protection Against Radiation?	☐	☒
6.	**NOTE: Changes to the plant or method of evaluation that result in re-analysis of the FSAR loss-of-coolant accident (LOCA) analysis require a screening.** Does the activity require a change to the FSAR LOCA analysis results subject to 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors? If the answer is <u>YES</u>, process the applicable changes per NP 5.2.12, 10 CFR 50.46 Reporting Requirements, and NP 5.2.6, FSAR Revisions.	☐	☒
7.	**NOTE: Regulatory commitments are found in the CLB Commitment Database.** Does the activity involve a change to a Regulatory Commitment ? If the answer is <u>YES</u>, process the applicable changes per NP 5.1.7, Regulatory Commitment Changes.	☐	☒

Point Beach Nuclear Plant

10 CFR 50.59/72.48 APPLICABILITY FORM

	Regulatory or Plant Process	YES	NO
8.	Does the activity involve a change to the Environmental Manual (EM), Radiological Effluent Control Program Manual (RECM), Offsite Dose Calculation Manual (ODCM), or Process Control Program (PCP), AND does NOT involve changes in use of explosive gases in waste treatment systems? If the answer is YES, document the applicable changes per the requirements of TS 5.5.1.	☐	☒
	NOTE: For purposes of determining 10 CFR 50.59 / 72.48 applicability, the determination of an administrative procedure below takes precedence over definitions or classifications in other plant procedures or guidelines.		
9.	Does the activity involve initial issue or a change to an administrative procedure or controlled document, OR an administrative software change ONLY? ALL of the following statements have to be marked YES for the procedure, controlled document, or software change to be considered administrative.	☒	☐
	a. **DOES NOT** direct, control or provide data as to how plant structures, systems, or components are operated, maintained, tested or repaired either specifically **OR** generically.	☒	☐
	b. **DOES NOT** specify acceptance criteria or operating limits for plant structures, systems, or components or demonstrates criteria/limits are met.	☒	☐
	c. **DOES NOT** specify parts, materials, chemicals, lubricants, etc. to be used in plant structures, systems, or components.	☒	☐
	d. **DOES NOT** specify compensatory action(s) to address plant structures, systems, or components out of service, or to address non-conforming conditions.	☒	☐
	e. **DOES NOT** affect operator access to operating areas of the plant.	☒	☐

10 CFR 50.59/72.48 APPLICABILITY CONCLUSION

NOTE: If ANY portion of the activity is NOT controlled by one or more of the processes above, further 10 CFR 50.59 / 72.48 review is required (i.e., portions not covered by the above processes shall be prescreened to other criteria or screened).

	YES	NO
ALL aspects of the activity are controlled by one or more of the processes above, therefore NO additional 10 CFR 50.59 and 72.48 review is required.	☒	☐

If the above question is answered NO, briefly describe the portions of the activity NOT covered by one or more of the above processes:

N/A

	Name (Print)	Signature	Date
Performed By	PATRICK J. Schwartz	Patrick J Schwartz	5/3/04
Reviewed By	Todd M. Gemskie	T M Gemskie	5/3/04

Point Beach Nuclear Plant

PROCEDURE PREPARER/TECHNICAL REVIEWER CHECKLIST

PART I- Preparer Checklist (page 1 of 2)

Procedure Number **EP Appendix B** Revision **22** Unit **PB0**

Title **Emergency Classification**

	Review Requirements	YES	N/A
1.	Ensure the procedure purpose is clearly stated and the procedure accomplishes the purpose.	☐	☒
2.	Ensure necessary precautions, prerequisites and controls are included to address potential industrial safety hazards and radiation safety hazards such as personnel protection, hazardous materials, waste or environments. Consult Industrial Health and Safety or Radiation Protection as necessary.	☐	☒
3.	Ensure responsibilities and qualifications for procedure performance are clearly defined (e.g., licensed SRO, NDE Level III, VT-2 examiner, certified QC inspector, position responsible for making acceptance criteria determinations, etc.).	☐	☒
4.	Ensure the procedure complies with applicable writer's guide (PBNP Site guide or AOP/EOP guide) for format and content requirements and is written in a manner that is easily followed and understandable.	☐	☒
5.	Ensure action steps are written as short and concise sentences and the number of actions in each step is limited to one, unless the actions are functionally related, AND the actions can be performed at the same time.	☐	☒
6.	Ensure steps return equipment to original or desired condition, adequately address post maintenance testing requirements and, specify any required follow-up actions to maintain status control, appropriate plant configuration, etc.	☐	☒
7.	Ensure references identified in the procedure are correct (e.g. correct Unit, train, set points, etc. Procedures and other documents being referenced are correctly identified, not cancelled, etc.).	☐	☒
8.	Ensure the procedure complies with the references, including codes, standards, design bases documents, drawings, vendor manuals and other documents (review applicable documents or data sources as necessary).	☐	☒
9.	Ensure the procedure complies with Current or ISFSI Licensing Basis requirements (as defined in NP 5.1.8 and NP 5.1.7) such as Tech Specs, FSAR, and will not cause any violations or discrepancies such as involuntary Tech Spec Action Conditions, etc. (review applicable documents or data sources as necessary).	☐	☒
10.	Ensure all regulatory commitments and Quality Assurance requirements are incorporated by reviewing the applicable documents or data sources.	☐	☒

PART I- Preparer Checklist (page 2 of 2)

	Review Requirements	YES	N/A
11.	Ensure the procedure addresses applicable design, vendor, scientific and technical information (review information as appropriate).	☐	☒
12.	Ensure the procedure adequately addresses related program requirements [e.g., Reactivity Management (NP 7.6.7), Foreign Materials Exclusion (NP 8.4.10), Infrequently Performed Tests or Evolutions (NP 1.2.6), Temporary Mods (NP 7.3.1, etc.]. (B-1, B-2, B-10). Review documents as necessary.	☐	☒
13.	Ensure appropriate Configuration Management has been addressed (e.g., impact on other equipment, documents, software or simulator. Operator aids identified, equipment names and numbers agree with Plant equipment and other documents or data sources, etc.). Ensure necessary changes are initiated.	☐	☒
14.	Ensure internal and industry operating experience has been addressed in the procedure by reviewing related action requests, other applicable "t-Track" items, INPO, NRC, EPRI documents, etc.	☐	☒
15.	Ensure required forms are completed and attached (e.g., Infrequently Performed Tests or Evolutions, Temporary Mods, CHAMPS callups and equipment record sheets processed, etc.). (B-1, B-2, B-9, B-10)	☐	☒

Comments

All boxes N/A. Procedure is strictly a chart. Procedures EPIP 1.2, Emergency Classification, and EPIP 1.2.1, Emergency Action Levels Technical Basis, have also been revised and address these issues directly.

Performed By PATRICK J. SCHWARTZ / [signature] Date 5-13/04
(Procedure Writer)

Point Beach Nuclear Plant

PROCEDURE PREPARER/TECHNICAL REVIEWER CHECKLIST

PART II- Technical Reviewer Checklist (page 1 of 2)

Note: The Technical Reviewer may also use or review the questions in Part I to aid review. For revisions not considered total rewrites, consider the checklist questions for the changes being made (at minimum) and the effect of the changes on the procedure.

Procedure Number **EP Appendix B** Revision **22** Unit **PB0**

Title **Emergency Classification**

	Review Requirements	YES	NO	N/A
1.	Ensure necessary precautions, prerequisites and controls are included to address potential industrial safety hazards and radiation safety hazards such as personnel protection, hazardous materials, waste or environments. Consult Industrial Health and Safety or Radiation Protection as necessary.	☐	☐	☒
2.	Ensure the procedure contains all limitations and cautions required to protect personnel or prevent equipment damage, including equipment limitations or interference with other equipment requirements such as EQ or Appendix R requirements.	☐	☐	☒
3.	Ensure the procedure is technically correct and the performance methods (flow paths, testing methodology, maintenance methods, operating sequence, etc.) are correct to preclude unexpected operability issues and maintain appropriate plant configuration (perform walk-downs as appropriate and review documents such as drawings, vendor manuals, interfacing procedures, etc.).	☐	☐	☒
4.	Ensure appropriate contingency actions are addressed such as appropriate actions for unacceptable procedure results, acceptance criteria, etc.	☐	☐	☒
5.	For procedure changes that manipulate equipment (valves, breakers, switches, locks, etc.), ensure restoration lineups and independent verification are provided as required, OR procedures are referenced that provide required restoration and independent verification. (B-6, B-7)	☐	☐	☒
6.	Ensure procedure steps are written to avoid preconditioning of equipment (e.g., avoid actions that could influence test results). (B-8)	☐	☐	☒
7.	Ensure steps that require, or may require, entry into a Tech Spec Action Condition are identified (review Tech Specs, etc. as necessary).	☐	☐	☒
8.	Ensure hold point and independent verification processes have been appropriately used in the procedure (reference NP 8.4.1, NP 2.1.2). (B-5)	☐	☐	☒

PROCEDURE PREPARER/TECHNICAL REVIEWER CHECKLIST

PART II- Technical Reviewer Checklist (page 2 of 2)

	Review Requirements	YES	NO	N/A
9.	Ensure qualitative and quantitative acceptance criteria values clearly indicate acceptable values and tolerances, the values are correct, and consistent with requirements such as Tech Specs, ASME codes, etc. Review documents as necessary.	☐	☐	☒
10.	Ensure the procedure complies with the references, including codes, standards, design bases documents, drawings, vendor manuals and other documents (review applicable documents or data sources as necessary).	☐	☐	☒
11.	Ensure the procedure complies with Current or ISFSI Licensing Basis requirements (as defined in NP 5.1.8 and NP 5.1.7) such as Tech Specs, FSAR, and will not cause any violations or discrepancies such as involuntary Tech Spec Action Conditions, etc. (review applicable documents or data sources as necessary).	☐	☐	☒

Comments (Explain any "NO" Answers):

All boxes N/A. Procedure is strictly a chart. Procedures EPIP 1.2, Emergency Classification, and EPIP 1.2.1, Emergency Action Levels Basis, have also been revised and address these issues directly.

Performed By Todd M. Gemskie / Todd Gemskie (Technical Reviewer) Date 5/03/04

Point Beach Nuclear Plant

10 CFR 50.54(q) EVALUATION CHECKLIST

Document EP Appendix B Title Emergency Classification Rev 22

1. Describe change:

Revise chart from NUREG 0654 to NEI 99-01 scheme.

2. Indicate which of the following standards of 10 CFR 50.47(b) may be affected by the change.

A. Assignment of Responsibility	☐
B. Onsite Emergency Organization	☐
C. Emergency Response Support and Resources	☐
D. Emergency Classification System	☒
E. Notification Methods and Procedures	☐
F. Emergency Communications	☐
G. Public Education and Information	☐
H. Emergency Facilities and Equipment	☐
I. Accident Assessment	☒
J. Protective Response	☐
K. Radiological Exposure Control	☐
L. Medical and Public Health Support	☐
M. Recovery and Reentry Planning and Post-accident Operations	☐
N. Exercises and Drills	☐
O. Radiological Emergency Response Training	☐
P. Responsibilities for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plan	☐

Does the change/revision result in the loss of ability to meet any of the standards of 10 CFR 50.47(b) or any NRC approved alternatives to those standards? ☐ YES ☒ NO
(Specifically discuss any item checked above, explaining why the standard is, or is not, being met).

D. Emergency Classification System - Convert from NUREG 0654 to NEI 99-01 Rev 4

I. Accident Assessment - Considers mode applicability in the new scheme.

3. Indicate which of the following areas of 10 CFR 50, Appendix E may be affected by the change.	
(i)(ii)(iii) Emergency plan as described in the FSAR	☐
(iv) A. Organization for coping with radiological emergencies	☐
(iv) B. Assessment of radiological emergencies	☐
(iv) C. Classifications, EALs and ERO Activation	☒
(iv) D. Notification of Federal, State and local agencies and the public	☐
(iv) E. ERFs, equipment, and communications	☐
(iv) F. Training, drills, and exercises	☐
(iv) G. Plans and procedures and surveillance of equipment and supplies	☐
(iv) H. Re-entry and Recovery following an accident	☐
(v) Emergency Response Data System (ERDS)	☐

Does the change/revision result in the loss of ability to meet any of the requirements of 10 CFR 50, Appendix E or any NRC approved alternatives to those requirements? ☐ YES ☒ NO
(Specifically discuss any item checked above, explaining why the requirement is, or is not, being met).

(IV) C. New scheme - leaving NUREG 0654, going to NEI 99-04 Rev 4.

4. Does the change/revision result in a reduction of any commitment that is not justified by the basis for that commitment? ☐ YES ☒ NO

Justification for Answer:

RG 1.101 Rev 4 accepts the new scheme of EALs (99-01 Rev 4).

5. Does the change/revision decrease the effectiveness of the emergency plan (including EALs)? ☒ YES ☐ NO

Justification for Answer:
(Explain how the change / revision will not reduce the capability or resources for emergency response or explain why the change/revision will result in a commensurate reduction in the need for such capabilities and resources.)

New scheme addresses mode applicability and may result in a different classification.

6. 50.54(q) Evaluation

______ Proposed change DOES NOT require prior NRC approval (all questions above answered "No")
__X__ Proposed change DOES require prior NRC approval (any question above answered "Yes")

Prepared by: Patrick J. Schwartz / Patrick J. Schwartz Date: 5/3/04
Reviewed by: James W. Connolly / James W. Connolly Date: 5/3/04
Regulatory Affairs
Approved by: Monica Ray / Monica Ray Date: 5/3/04
EP Manager

	GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
	[illegible]	None	None	MU1.1 An unplanned sustained positive startup rate observed on nuclear instrumentation
	[illegible]	None	None	MU2.1 Coolant activity ≥ 0.8 μCi/gm dose equivalent I-131
	None	None	None	MU3.1 Failed Fuel Monitor (RE-109) ≥ 24 mrem/hr not due to a planned evolution
Reactor Fuel	None	None	MA4.1 Confirmed sustained (10 min. average) alarm on any of the following radiation monitors resulting from an uncontrolled fuel handling process indicating damaged or uncovered irradiated fuel: - RE-105 SFP Area Low Range Area Radiation High Alarm (≥ 10 mR/hr) - RE-135 SFP Area High Range Area Radiation High Alarm (≥ 100 mR/hr) - 1(2) RE-211 Containment Air Particulate Monitor High Alarm (≥ 0.5 μCi) - 1(2) RE-212B Containment Background Monitor High Alarm (≥ 100 mR/hr) MA4.2 Report of visual observation of irradiated fuel uncovered OR Loss of refueling water inventory as indicated by excessive makeup rate or unexpected lowering in refueling water storage tank level	MU4.1 Spent fuel pool (reactor cavity during refueling) water level cannot be restored and maintained above the spent fuel pool low water level alarm setpoint AND Unplanned SFP Area Radiation Monitor reading rise: - RE-105 SFP Area Low Range Area Radiation Monitor - RE-135 SFP Area High Range Area Radiation Monitor
RCS Leakage	None	None	None	MU6.1 Unidentified or pressure boundary leakage ≥ 10 gpm OR Identified leakage ≥ 25 gpm
Comm. Loss	None	None	None	MU8.1 Loss of all communications capability affecting the ability to EITHER: Perform routine operations OR Notify offsite agencies or personnel
Radioactivity Release / Area Radiation	RG1.1 A valid reading on any monitors Table R-1 column "GE" for ≥ 15 min. unless dose assessment can confirm releases are below Table R-2 column "GE" within this time period	RS1.1 A valid reading on any monitors Table R-1 column "SE" for ≥ 15 min. unless dose assessment can confirm releases are below Table R-2 column "SE" within this time period	RA1.1 Loss of control of radioactive materials as indicated by a valid reading on any monitors Table R-1 column "Alert" for ≥ 15 min. unless sample analysis can confirm release rates ≤ 200 x ODCM limits within this time period	RU1.1 Loss of control of radioactive materials as indicated by a valid reading on any monitors Table R-1 column "UE" for ≥ 60 min. unless sample analysis can confirm release rates ≤ 2 x ODCM limits within this time period
	RG2.1 Dose projections or field surveys resulting from actual or imminent release which indicate doses / closed-window dose rates ≥ Table R-2 column "GE" at the site boundary or beyond	RS2.1 Dose projections or field surveys resulting from actual or imminent release which indicate doses / dose rates ≥ Table R-2 column "SE" at the site boundary or beyond	RA2.1 An unplanned gaseous or liquid release with rates ≥ 200 x ODCM limits for ≥ 15 min. RA2.2 Dose projections or field surveys resulting from an unplanned actual or imminent release which indicate doses / dose rates ≥ Table R-2 column "Alert" at the site boundary or beyond	RU2.1 An unplanned gaseous or liquid release with rates ≥ 2 x ODCM limits for ≥ 60 min.
	None	None	RA3.1 Sustained (10 min. average) area radiation levels ≥ 15 mR/hr in EITHER: Control Room (RE 101) OR Central Alarm Station (by survey) OR Secondary Alarm Station (by survey) RA3.2 Sustained (10 min. average) abnormal area radiation levels ≥ 12 R/hr in any Table H-1 Safe Shutdown Area AND Access to affected area is required for safe operation or shutdown	RU3.1 Any sustained (10 min. average) direct ARM readings ≥ 100 x alarm or offscale high not resulting from a planned evolution
Hazards	HG1.1 An adversary has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions: - Reactivity control - RCS inventory - Secondary heat removal - Spent Fuel Pool integrity	HS1.1 Intrusion into a Vital Area by an adversary indicated by notification by the Security Shift Supervisor to implement AOP-29 for Vital Area intrusion	HA1.1 Intrusion into the site Protected Area by an adversary indicated by notification by the Security Shift Supervisor to implement AOP-29 for a PA intrusion	HU1.1 Indication of attempted sabotage, hostage/extortion, civil disturbance or strike action onsite OR Notification of any credible site-specific LOW threat by the Security Shift Supervisor or outside agency (NRC, military or law enforcement)
	None	None	HA2.1 Fire or explosion in any Table H-1 Safe Shutdown Area, which results in EITHER: Visible damage to plant equipment or structures needed for safe shutdown OR Affected safety system performance is degraded indicating damage to a safety system	HU2.1 Confirmed fire in the Protected Area requiring offsite assistance HU2.2 Report by plant personnel of an explosion within Protected Area resulting in visible damage to permanent structures or equipment
	None	None	HA3.1 Vehicle crash which precludes personnel access to or damages equipment in one or more Table H-1 Safe Shutdown Areas HA3.2 Report or detection of toxic or flammable gases within any Table H-1 Safe Shutdown Area in concentrations that EITHER: Will be immediately life threatening to plant personnel (IDHL) OR Exceed the lower flammability limit	HU3.1 Vehicle crash into plant structures or systems within the Protected Area HU3.2 Report or detection of toxic or flammable gases that could enter or have entered within the Protected Area in amounts that could affect the health of plant personnel or safe plant operation OR Report by local, county or state officials for evacuation or sheltering of site personnel based on an offsite event
	None	None	HA4.1 Two or more seismic monitors (SEI 6210 through 6213) indicate ground acceleration EITHER: ≥ 0.08 g horizontal OR ≥ 0.04 g vertical HA4.2 Sustained (15 min. average) winds > 105 mph onsite OR Tornado strikes any Table H-1 Safe Shutdown Area HA4.3 Uncontrolled flooding that results in degraded safety system performance or that creates industrial safety hazards that precludes access necessary to operate or monitor safety equipment in EITHER: - The auxiliary building caused by rupture of the SW header OR - The water intake structure caused by rupture of a circulating water system expansion joint or fire water main	HU4.1 Actuation of 2 or more seismic monitors (SEI 6210 through 6213) AND Verified by: - Actual ground shaking OR - By contacting the U.S. Geological Survey National Earthquake Information Center HU4.2 Sustained (15 min. average) winds > 75 mph onsite OR Report by plant personnel of tornado striking within plant Protected Area HU4.3 Uncontrolled flooding in the auxiliary building caused by rupture of the SW header OR Uncontrolled flooding in the water intake structure caused by rupture of a circulating water system expansion joint or fire water main
Control Rm Evacuation	None	HS5.1 Control Room evacuation AND Transfer of reactivity, RCS inventory and secondary heat removal control functions cannot be established per AOP-10A Safe Shutdown - Local Control in ≤ 15 min.	HA5.1 Entry into AOP-10 Control Room Inaccessibility due to Control Room Evacuation	None
Other	HG6.1 Any event in the judgment of the Emergency Director is in progress which indicates actual or imminent core damage and the potential for a large release of radioactive material in excess of EPA Protective Action Guides outside the site boundary	HS6.1 Any event in the judgment of the Emergency Director is in progress which indicate actual or likely failures of plant systems needed to protect the public. Any releases are not expected to result in exposures which exceed EPA Protective Action Guides beyond the site boundary	HA6.1 Any event in the judgment of the Emergency Director, that could cause or has caused actual substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of EPA Protective Action Guides	HU6.1 Any event, in the judgment of the Emergency Director, that could lead to or has led to a potential degradation of the level of safety of the plant
ISFSI Events	None	None	None	IU1.1 Loss of cask confinement boundary as indicated by exceeding any of the following external surface dose rates on any loaded Dry Storage Cask: - ≥ 100 mR/hr at the cask side - ≥ 200 mR/hr at the top of the cask - ≥ 350 mR/hr at the cask air inlet - ≥ 100 mR/hr at the cask air outlet
	None	None	None	IU1.2 Report by Security Shift Supervisor of a security concern within the ISFSI

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	GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
ATWS	[illegible] entry into Subcriticality-RED path [illegible] AND Conditions requiring entry into EITHER: Core Cooling-RED path (CSP-C.1) OR Heat Sink-RED path (CSP-H.1)	MS7.1 Conditions requiring entry into Critical Safety Function - Subcriticality-RED path (CSP-S.1)	MA7.1 Any failure of the Reactor Protection System to generate an automatic trip signal and reduce power range to ≤ 5%	None
Loss of AC Power Sources	MG8.1 Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 AND EITHER: Power restoration to any safety-related 4160 VAC bus or 480 VAC bus is not likely in ≤ 4 hours OR Conditions require entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)	MS8.1 Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥ 15 min.	MA8.1 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 ≥ 15 min. (one source away from station blackout): - A single emergency diesel generator (G01, G02, G03 or G04) - LVSAT 1(2)-X04 - UAT 1(2)-X02 - Cross-tying with the opposite unit power supply	MU8.1 Unplanned loss of offsite AC power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥ 15 min.
Loss of DC Power Sources	None	MS9.1 ≤ 105 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for ≥ 15 min. due to unplanned activities	None	None
Equipment Failures	None	None	None	MU10.1 Plant is not brought to required operating mode within Technical Specifications LCO required action completion time
	None	None	MA11.1 Turbine failure generated missiles resulting in visible damage to or penetrating any Table H-1 Safe Shutdown Area structure or system	MU11.1 Report of main turbine failure requiring turbine trip resulting in: Damage to turbine-generator seals OR Casing penetration
	None	MS12.1 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥ 15 min. AND PPCS is unavailable AND Complete loss of ability to monitor all critical safety function status AND A significant transient is in progress	MA12.1 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥ 15 min. AND EITHER: A significant transient is in progress OR PPCS is unavailable	MU12.1 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥ 15 min.
Fission Product Barriers	FG1.1 Loss of any two barriers with a loss or challenge of a third (Table F-1)	FS1.1 Loss or challenge of any two barriers (Table F-1)	FA1.1 Any loss or challenge of Fuel Cladding or RCS (Table F-1)	FU1.1 Any loss or challenge of Containment (Table F-1)

Table F-1 FISSION PRODUCT BARRIER MATRIX

Fuel Clad Barrier		RCS Barrier		Primary Containment Barrier	
☐ Loss	☐ Challenge	☐ Loss	☐ Challenge	☐ Loss	☐ Challenge
1. Conditions requiring entry into Core Cooling-RED path (CSP-C.1) 2. Coolant activity ≥ 300 μCi/gm I-131 equivalent 3. CET readings ≥ 1200°F (core cooling red path CSP-C.1) 4. Containment rad monitor reading ≥ 17 R/hr 5. Failed Fuel Monitor (RE-109) reading ≥ 120 mR/hr 6. Emergency Director judgment	1. Conditions requiring entry into Core Cooling-ORANGE path (CSP-C.2) 2. Conditions requiring entry into Heat Sink-RED path (CSP-H.1) 3. CET readings ≥ 700°F 4. RVLIS NR ≤ 38% with no RCPs running 5. Emergency Director judgment	1. RCS subcooling based on core exit thermocouples ≤ [illegible] due to RCS leakage 2. SGTR in excess of available charging pumps 3. Containment rad monitor reading ≥ 4.0 R/hr 4. Emergency Director judgment	1. Conditions requiring entry into RCS Integrity-RED path (CSP-P.1) 2. Conditions requiring entry into Heat Sink-RED path (CSP-H.1) 3. Unisolable leak exceeding 50 gpm 4. Emergency Director judgment	1. Rapid unexplained containment pressure drop following initial rise 2. Containment pressure or sump level response not consistent with LOCA conditions 3. Ruptured S/G is also faulted outside of containment 4. Primary-to-secondary leakage ≥ 10 gpm with non-isolable steam release from affected S/G to the environment 5. Containment isolation required and containment isolation or ventilation valve(s) not closed when required AND Radiological release pathway to the environment exists 6. Inability to isolate any primary system discharging outside containment AND Radiological release pathway to the environment exists 7. Emergency Director judgment	1. Conditions requiring entry into Containment-RED path (CSP-Z.1) 2. Containment pressure ≥ 60 psig and rising (Containment RED path, CSP-Z.1) 3. Hydrogen concentration in containment ≥ 6% 4. Containment pressure ≥ 25 psig with less than one train of containment spray and two containment accident fan coolers operating 5. CET readings ≥ 1200°F (Core Cooling-RED path, CSP-C.1) AND Restoration procedures not effective within 15 min. 6. CET > 700°F with RVLIS NR < 38% and no RCPs running (Core Cooling-RED path, CSP-C.1) AND Restoration procedures not effective within 15 min. 7. Containment rad monitor reading ≥ 15,000 R/hr 8. Emergency Director judgment

Table R-1 Effluent Monitor Classification Thresholds				
Monitor	GE	SE	Alert	UE
1(2)RE 212	—	—	5.46E-3 μCi/cc	5.46E-4 μCi/cc
1(2)RE 305	—	—	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 307	1.44E1 μCi/cc	1.44E0 μCi/cc	—	—
1(2)RE 309	1.44E1 μCi/cc	1.44E0 μCi/cc	—	—
RE 214	—	—	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 216	—	—	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 317	2.40E0 μCi/cc	2.40E-1 μCi/cc	—	—
RE 319	2.40E0 μCi/cc	2.40E-1 μCi/cc	—	—
1(2)RE 215	—	—	5.42E+2 μCi/cc	5.42E0 μCi/cc
RE 218	—	—	3.72E+2 μCi/cc	3.72E0 μCi/cc
RE 220	4.04E+3 μCi/cc	4.04E+2 μCi/cc	—	—
RE 221	—	—	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 223	—	—	6.16E-2 μCi/cc	2.16E-4 μCi/cc
RE 627	4.34E0 μCi/cc	4.34E-1 μCi/cc	—	—
RE 224	—	—	4.16E-1 μCi/cc	4.16E-4 μCi/cc
1(2)RE 231/232	—	—	—	—
1(2)SG A/B				
1 ARV	6.68E-1 μCi/cc	6.68E-2 μCi/cc	N/A	N/A
1 SRV	2.44E-1 μCi/cc	2.44E-2 μCi/cc	N/A	N/A
2 SRV	1.34E-1 μCi/cc	1.34E-2 μCi/cc	N/A	N/A
3 SRV	6.25E-3 μCi/cc	6.25E-4 μCi/cc	N/A	N/A
4 SRV	6.30E-2 μCi/cc	6.30E-3 μCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.88E-3 μCi/cc	5.88E-5 μCi/cc
RE 230*	N/A	N/A	7.40E-2 μCi/cc	7.40E-4 μCi/cc

* with Waste Water Effluent discharge not isolated

Table H-1 Safe Shutdown Areas
- 1(2) Containment Building
- Primary Auxiliary Building
- Turbine Building
- Control Building
- Diesel Generator Building
- Gas Turbine Building
- Circ Water Pumphouse

Table R-2 Dose Proj / Env. Measurement Class. Thresholds	GE	SE	Alert
TEDE	1000 mRem	100 mRem	10 mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (1 hr of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

Page 5 4-12-04
[illegible]

HOT CONDITIONS (RCS > 200°F)

Modes: 1 Power Operations · 2 Startup · 3 Hot Standby · 4 Hot Shutdown · 5 Cold Shutdown · 6 Refuel · D Defueled

~~EPIP 1.2 Emergency Classification~~

Approved ______ Manager Emergency Preparedness ______ Date

EP App B

GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
[illegible]	[illegible]	[illegible]	[illegible]

GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
[illegible]	[illegible]	[illegible]	[illegible]

[illegible]

COLD CONDITIONS (RCS ≤ 200°F)

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | D Defueled

[illegible]

EP App. B

WORD PROCESSING DOCUMENT CHECKLIST

Routing	Extension	Date	Action Or Comments

TEMPORARY CHANGE TO BE MADE PERMANENT? ☐ **YES** **TEMP CHANGE NO.** ________

DOC. ID: EP Appendix B **NEW REV:** 22 **RETRIEVE FROM:** ________	**DRAFT DUE DATE:** ________ **ISSUE DUE DATE:** ________

Comments for WP: ☐ **STAFF PROCEDURE**

Writing for Updated Electronic Copy.

INITIAL PROCESSING **DATE IN:**	4/15
Update **Master Index** with your name and special comments if necessary.	
Verify **TITLE, USAGE LEVEL, CLASSIFICATION** and **CURRENT REVISION** against master for *uncontrolled* procedures.	
Connect to Macros and run set-up macros: ▪ Options (shift alt ctrl O) ▪ TNR 12 (shift alt ctrl N) ▪ Non-breaking Hyphens (shift alt ctrl -) ▪ Remove all Rev Bars (shift alt ctrl R)	
Update Header/Footer: ▪ Up the Revision # and type DRAFT ▪ Usage Level ▪ Current Date ▪ Check classification in header with cover page and PBF-0026a/c. No classification for controlled reference documents and administrative procedures.	
Change MSS (Manager's Supervisory Staff) to PORC (Plant Operation's Review Committee)	
Temp Changes: Ensure all changes are incorporated by verifying them against PBF-0026c. (DO NOT change Approval Authority/Reviewer on Permanent Temp Changes.)	
Make appropriate changes and rev bar. (Rev 0's and Total Rewrites **DO NOT** get rev barred)	
Check bookmarks/cross-references.	
Ensure reference titles are correct and still effective. **(Not Including Temp Changes)**	
Rerun Non-breaking hyphen macro. Spell check document and turn hidden text off before printing.	
Print and proofread document: ▪ Page endings and step numbering are correct (including Table of Contents) ▪ Page Numbers ▪ Error References ▪ Attachments are included	
Save copy in EDMS, and in EDMS Backup. List here, if different.________	
Delete electronic copy in Draft Transfers after saving in EDMS Backup.	
For first time clean-ups: Attach yellow Conversion Verification Form.	Yes / No
Update Master Index with date forwarded to originator/Distribution.	

SUBSEQUENT REVISIONS **DATE IN:**					
Make appropriate changes and rev bar.					
Rerun Non-breaking hyphen macro. Spell check document and turn hidden text off before printing.					
Remove hidden text, print and proofread document.					
Save copy in EDMS and again in EDMS Backup.					
date Master Index.					
DATE OUT:					
INITIALS:					

Date Returned for Issue:

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Point Beach Nuclear Plant

PROCEDURE VALIDATION

Procedure Number **Emergency Plan, Appendix B** Revision ____ Unit **PB0**

Title **Emergency Action Level (EAL) Overview Matrix**

METHOD OF VALIDATION

(Methods may be combined when necessary)

☐ Walk-down (normally used unless plant conditions prohibit or ALARA concerns are involved)

☒ Simulator ☐ Simulation ☒ User group review ☒ Other Operations Support Services, Inc.

REVIEW REQUIREMENTS		YES	NO	N/A
1.	Ensure the procedure accomplishes the stated purpose without introducing new challenges to personnel or equipment (e.g., no reactivity issues or involuntary entry to Current Tech Spec LCO or Improved Tech Spec Action Condition. Proper flow paths, maintenance, or testing methods identified).	☒	☐	☐
2.	Ensure the procedure contains all required processes and steps and can be performed as written and in sequence written.	☒	☐	☐
3.	Ensure the procedure is written such that qualified users should be able to perform it consistently.	☒	☐	☐
4.	Ensure Industrial and Radiological Safety have been appropriately addressed for personnel protection.	☒	☐	☐
5.	Ensure the procedure effectively addresses the communication and coordination necessary for procedure performance, both internal and external to responsible organization.	☒	☐	☐

Explain "NO" Answers:

(Use additional sheets as necessary) (Continued) ☐

N/A

Comments and Suggestions for Improvement:

(Use additional sheets as necessary) (Continued) ☐

The validation process ensured that the Emergency Action Levels (EALs) were usable and operationally correct. The validation process also identified the ability of validation team members to arrive at consistent interpretations of EALs under varying conditions. The validation team was led by Emergency Planning, supported by Operations Support Services, Inc., and included various site disciplines from operations, engineering, emergency planning, security and operations training. The test methodology was comprised of table-top and control room simulator scenarios requiring the validation team to conduct step-by-step user actions to implement classification of selected EALs. Validation team participants are listed on the attached PBNP Training Attendance Report.

Performed By: **Monica Ray** Monica Ray Date 5/10/04

(Validator)

Point Be[illegible] Nuclear Plant

TRAINING ATTENDANCE REPORT

Activity Code:	Lesson Title	Rev	Initial/ Cont	Date(s)	Hrs	Evaluation ID
1.	OSSI - TABLETOP DRAFT EAL/Simulator use	NA		12/17/03	8.0	
2.	for EAL classification capability.					
3.						
4.						

Trainee's Name *Please print name legibly*	Trainee's Signature	Trainee's ID WE = WE Employee ID # WPS Union = WPS ID # Contractor = Social Sec. # C# = NMC Train ID #	Activity Status / Grade 1 Status	1 Grade	2 Status	2 Grade	3 Status	3 Grade	4 Status
1. Patrick J. Schwartz	Patrick J Schwartz	XX7215							
2. James E Schleif	James E Schleif	WE4809							
3. Steve Boque	S Boque	PB4807							
4. Charles Strazer		W33913							
5. Jim Green	J Green	WE3333							
6. Tim Lothe		WE5462							
7. Tim Taylor		WE6307							
8. Rick Merkes		PB1539							
9. Daniel C. Pond	Daniel C Pond	PB3143							
10.									

Instructor's Name *(Please print)*	Instructor's Signature	SME*	Instructor's ID	1	2	3
Kelly Walker						
Edward G. Bates						

Comments: M C DAVIS

PBF-6401
Revision 10 07/31/03
File: A8.16

FRONT

Status values: P (Successfully completed activity requirements)
F (Failed activity requirements)
A (Attended without evaluation)
I (lesson incomplete – makeup required)
T (Taught – Registrar entry desired) *use "X" if Reg. entry not wanted*
X (No registrar entry desired – e.g., trainee did not attend activity)

Grade values:

* Check mark SME column if instructor is an SME (versus a Training instructor)

EMERGENCY ACTION LEVEL (EAL) OVERVIEW MATRIX

		GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
Reactor Fuel	Inadvertent Criticality	NONE	NONE	NONE	MU1.1 1 3 4 5 6 An unplanned sustained positive startup rate observed on nuclear instrumentation
Reactor Fuel	Coolant Activity	NONE	NONE	NONE	MU2.1 1 2 3 4 5 6 Coolant activity ≥0.8 µCi/gm dose equivalent I-131
Reactor Fuel	Failed Fuel Monitor	NONE	NONE	NONE	MU3.1 1 2 3 4 Failed Fuel Monitor (RE-109) ≥24 mrem/hr not due to a planned evolution
Reactor Fuel	Refueling Accidents & Other Radiation Monitors	NONE	NONE	MA4.1 1 2 3 4 5 6 D Confirmed sustained (10 min average) alarm on any of the following radiation monitors resulting from an uncontrolled fuel handling process indicating damaged or uncovered irradiated fuel: - RE-105 SFP Area Low Range Area Radiation High Alarm (≥10 mR/hr) - RE-135 SFP Area High Range Area Radiation High Alarm (≥100 mR/hr) - 1(2) RE-211 Containment Air Particulate Monitor High Alarm (≥0.5 µCi) - 1(2) RE-212B Containment Background Monitor High Alarm (≥100 mR/hr) MA4.2 1 2 3 4 5 6 D Report of visual observation of irradiated fuel uncovered **OR** *Loss of refueling water inventory as indicated by* excessive makeup rate or unexpected lowering in refueling water storage tank level	MU4.1 1 2 3 4 5 6 D Spent fuel pool (reactor cavity during refueling) water level cannot be restored and maintained above the spent fuel pool water level alarm setpoint **AND** Unplanned SFP Area Radiation Monitor readings rise - RE-105 SFP Area Low Range Area Radiation Monitor - RE-135 SFP Area High Range Area Radiation Monitor
RCS Leakage		NONE	NONE	NONE	MU5.1 1 2 3 4 5 Unidentified or pressure boundary leakage ≥ 10 gpm **OR** Identified leakage ≥ 25 gpm
Comm Loss		NONE	NONE	NONE	MU6.1 1 2 3 4 5 6 D Loss of all communications capability affecting the ability to **EITHER** Perform routine operations **OR** Notify offsite agencies or personnel
Radioactivity Release/Area Radiation	Effluent Monitors	RG1.1 1 2 3 4 5 6 D *A valid reading on any monitors* Table R-1 column "GE" for ≥15 min. unless dose assessment can confirm releases are below Table R-2 column "GE" within this time period	RS1.1 1 2 3 4 5 6 D A valid reading on any monitors Table R-1 column "SE" for ≥15 min. unless dose assessment can confirm releases are below Table R-2 column "SE" within this time period	RA1.1 1 2 3 4 5 6 D Loss of control of radiatioactive materials as indicated by a valid reading on any monitors Table R-1 column "ALERT" for ≥15 min. unless sample analysis can confirm release rates ≤200 x ODCM limits within this time period	RU1.1 1 2 3 4 5 6 D *Loss of control of radiatioactive materials* as indicated by a valid reading on any monitors Table R-1 column "UE" for ≥60 min. unless sample analysis can confirm release rates ≤2 x ODCM limits within this time period
Radioactivity Release/Area Radiation	Dose Proj./Env. Meas./Release Rates	RG2.1 1 2 3 4 5 6 D Dose projections or field surveys resulting from actrual or imminent release which indicate doses / closed -window dose rates ≥Table R-2 column "GE" at the site boundary or beyond.	RS2.1 1 2 3 4 5 6 D Dose projections or field surveys resulting from actual or imminent release which indicate doses / dose rates ≥Table R-2 column "SE" at the site boundary or beyond.	RA2.1 1 2 3 4 5 6 D An unplanned gaseous or liquid release with *rates ≥200 x ODCM limits for ≥15 min* RA2.2 1 2 3 4 5 6 D Dose projections or field surveys resulting from an unplanned actual or imminent release which indicate doses / dose rates ≥Table R-2 column "Alert" at the site boundary or beyond.	RU2.1 1 2 3 4 5 6 D An unplanned gaseous or liquid release with rates ≥2 x ODCM limits for ≥60 min.
Radioactivity Release/Area Radiation	Area Radiation Levels	NONE	NONE	RA3.1 1 2 3 4 5 6 D Sustained (10 min. average) area radiation levels ≥15 mR/hr in **EITHER** Control Room (RE 101) **OR** Central Alarm Station (by survey) **OR** Secondary Alarm Station (by survey) RA3.2 1 2 3 4 5 6 D Sustained (10 min. average) abnormal area radiation levels ≥12 *R/hr in any Table H-1 Safe Shutdown Area* **AND** Access to affected area is required for safe operation or shutdown.	RU3.1 1 2 3 4 5 6 D Any sustained (10 min. average) direct ARM readings ≥100 x alarm or offscale high not resulting from a planned event or evolution
Hazards	Security Threats	HG1.1 1 2 3 4 5 6 D An adversary has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions: - Reactivity control - RCS inventory - Secondary heat removal - Spent Fuel Pool integrity	HS1.1 1 2 3 4 5 6 D Intrusion into a Vital Area by an adversary indicated by notification by the Security Shift Supervisor to implement AOP-29 for Vital Area intrusion.	HA1.1 1 2 3 4 5 6 D Intrusion into the site Protected Area by an adversary indicated by notification by the Security Shift Supervisor to implement AOP-29 for a PA intrusion.	HU1.1 1 2 3 4 5 6 D Indication of attempted sabotage, hostage/extortion, civil disturbance or strike action onsite **OR** Notification of any credible site-specific threat by the Security Shift Supervisor or outside agency (NRC, military or law enforcement)
Hazards	Fire or Explosion	NONE	NONE	HA2.1 1 2 3 4 5 6 D Fire or explosion in any Table H-1 Safe Shutdown Area, which results in **EITHER**: Visible damage to plant equipment or structures needed for safe shutdown **OR** Affected safety system performance is degraded indicating damage to a safety system	HU2.1 1 2 3 4 5 6 D Confirmed fire in the Protected Area not extinguished in ≤15 min. of Control Room Notification HU2.2 1 2 3 4 5 6 D Report by plant personnel of an explosion within Protected Area resulting in visible damage to permanent structures or equipment.
Hazards	Vehicle Crash/Toxic and Flammable Gas	NONE	NONE	HA3.1 1 2 3 4 5 6 D Vehicle crash which precludes personnel access to or damages equipment in one or more Table H-1 Safe Shutdown Areas HA3.2 1 2 3 4 5 6 D Report or detection of toxic or flammable gases within any Table H-1 Safe Shutdown Area in concentrations that **EITHER**: Will be immediately life threatening to plant personnel (IDLH) **OR** Exceed the lower flammability limit	HU3.1 1 2 3 4 5 6 D Vehicle crash into plant structures or systems within the Protected Area HU3.2 1 2 3 4 5 6 D Report or detection of toxic or flammable gases that could enter or have entered within the Protected Area in amounts that could affect the health of plant personnel or safe plant operation **OR** Report by local, county or state officials for evacuation or sheltering of site personnel based on an offsite event
Hazards	Natural Events	NONE	NONE	HA4.1 1 2 3 4 5 6 D Two or more seismic monitors (SEI 6210 through 6213) indicate ground acceleration **EITHER**: ≥0.06 g horizontal **OR** ≥0.04 g vertical HA4.2 1 2 3 4 5 6 D Sustained (15 min. average) winds > *108 mph onsite* **OR** Tornado strikes any Table H-1 Safe Shutdown Area HA4.3 1 2 3 4 5 6 D Uncontrolled flooding that results in degraded safety system performance or that creates industrial safety hazards that precludes access necessary to operate or *monitor safety euipment in* **EITHER**: - The auxilliary building caused by rupture of the SW header **OR** - The water intake structure caused by rupture of a circulating water system expansion joint or fire water main	HU4.1 1 2 3 4 5 6 D Activation of 2 or more seismic monitors (SEI 6210 through 6213) **AND** Verified by: - Actual ground shaking **OR** - By contacting the U.S. Geological Survey National Earthquake Information Center HU4.2 1 2 3 4 5 6 D Sustained (15 min. average) winds > 75 mph onsite **OR** Report by plant personnel of tornado striking within plant Protected Area HU4.3 1 2 3 4 5 6 D Uncontrolled flooding in the auxiliary building caused by rupture of the SW header **OR** Uncontrolled flooding *in the water intake structure* caused by rupture of a circulating water system expansion joint or fire water main
Control Rm Evacuation		NONE	HS5.1 1 2 3 4 5 6 D Control Room evacuation **AND** Transfer of reactivity, RCS inventory and secondary heat removal control functions cannot be established per AOP-10A Safe Shutdown - Local Control in ≤15 min.	HA5.1 1 2 3 4 5 6 D Entry into AOP-10 Control Room Inaccessibility due to Control Room Evacuation	NONE
Other		HG6.1 1 2 3 4 5 6 D Any event in the judgement of the Emergency Director is in progress which indicates actual or imminent core damage and the potential for a large release of radioactive material in excess of EPA Protective Action Guides outside the site boundary	HS6.1 1 2 3 4 5 6 D Any event in the judgement of the Emergency Director is in progress which indicates actual or likely failures of plant systems needed to protect the public. Any releases are **not** expected to result in exposures which exceed EPA Protective Action Guides beyond the site boundary	HA6.1 1 2 3 4 5 6 D Any event in the judgement of the Emergency Director, that could cause or has caused actual substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of EPA Protective Action Guides	HU6.1 1 2 3 4 5 6 D Any event, in the judgement of the Emergency Director, that could lead to or has led to a potential degradation of the level of safety of the plant.
ISFSI Events	Loss of Cask Confinement	NONE	NONE	NONE	IU1.1 1 2 3 4 5 6 D Loss of cask confinement boundary as indicated by exceeding any of the following external surface dose rates on any loaded Dry Storage Cask: - ≥ 100 mR/hr at the cask side - ≥ 200 mR/hr at the top of the cask - ≥ 350 mR/hr at the cask air inlet - ≥ 100 mR/hr at the cask air outlet
ISFSI Events	Security	NONE	NONE	NONE	IU1.2 1 2 3 4 5 6 D Report by Security Shift Supervisor of a security concern within the ISFSI.

HOT CONDITIONS

	GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
ATWS	MG7.1 1 2 Conditions requiring entry into Subcritically-RED path (CSP-S.1) currently exist **AND** Conditions requiring entry into **EITHER**: Core Cooling-RED path (CSP-C.1) **OR** Heat Sink-RED path (CSP-H.1)	MS7.1 1 2 Conditions requiring entry into Critical Safety Function-Subcritically-RED path (CSP-S.1)	MA7.1 1 2 Any failure of the Reactor Protection System to generate an automatic trip signal and reduce power range to ≤5% **AND** Manual trip successful	NONE
Loss of AC Power Sources	MG8.1 1 2 3 4 Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 **AND EITHER:** Power restoration to any safety-related 4160 VAC bus or 480 VAC bus is not likely in ≤4 hours **OR** Conditions require entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)	MS8.1 1 2 3 4 Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥15 min.	MA8.1 1 2 3 4 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 ≥15 min. (one source away from station blackout): - A single emergency diesel generator (G01, G02, G03 & G04) - LVSAT 1(2)-X04 - UAT 1(2)-X02 - Cross-tying with the opposite unit power supply	MU8.1 1 2 3 4 Unplanned loss of offsite AC power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥15 min.
Loss of DC Power Sources	NONE	MS9.1 1 2 3 4 ≤105 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for ≥15 min. due to unplanned activities	NONE	NONE
Tech Spec	NONE	NONE	NONE	MU10.1 1 2 3 4 Plant is not brought to required operating mode within Technical Specifications LCO required action completion time
H.4 Equipment Failures – Turbine Failure	NONE	NONE	MA11.1 1 2 3 4 Turbine failure generated missiles resulting in visible damage to or penetrating any Table H-1 Safe Shutdown Area structure or system	MU11.1 1 2 3 4 Report of main turbine failure requiring turbine trip resulting in: Damage to turbine-generator seals **OR** Casing penetration
H.4 Equipment Failures – Loss of Indicators/Alarms	NONE	MS12.1 1 2 3 4 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 min. **AND** PPCS is unavailable **AND** Complete loss of ability to monitor all critical safety functions status **AND** A significant transient is in progress	MA12.1 1 2 3 4 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 min. **AND EITHER:** A significant transient is in progress **OR** PPCS is unavailable	MU12.1 1 2 3 4 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 min.
Fission Product Barriers	FG1.1 1 2 3 4 Loss of any two barriers with a loss or challenge of a third (Table F-1)	FS1.1 1 2 3 4 Loss or challenge of any two barriers (Table F-1)	FA1.1 1 2 3 4 Any loss or challenge of Fuel Cladding or RCS (Table F-1)	FU1.1 1 2 3 4 Any loss or challenge of Containment (Table F-1)

Table F-1 FISSION PRODUCT BARRIER MATRIX

Fuel Clad Barrier		RCS Barrier		Primary Containment Barrier	
☐ Loss	☐ Challenge	☐ Loss	☐ Challenge	☐ Loss	☐ Challenge
1. Conditions requiring entry into Core Cooling-RED path (CSP-C.1) 2. Coolant activity ≥300 uCi/gm I-131 equivalent 3. CET readings ≥1200°F 4. Containment rad monitor reading ≥17 R/hr 5. Failed Fuel Monitor (RE-109) reading ≥120 mRem/hr 6. Emergency Director judgment	1. Conditions requiring entry into Core Cooling-ORANGE path (CSP-C.2) 2. Conditions requiring entry into Heat Sink-RED path (CSP-H.1) 3. CET readings ≥700°F 4. RVLIS NR ≤25 ft with no RCPs running 5. Emergency Director judgment	1. RCS subcooling based on core exit thermocouples ≤[80°F] 35°F due to RCS leakage 2. SGTR in excess of available charging pumps 3. Containment rad monitor reading ≥3.0 R/hr 4. Emergency Director judgment	1. Conditions requiring entry into RCS Integrity-RED path (CSP-P.1) 2. Conditions requiring entry into Heat Sink-RED path (CSP-H.1) 3. Unisolable leak exceeding *60gpm* 4. Emergency Director judgment	1. Rapid unexplained containment pressure drop following initial rise 2. Containment pressure or sump level response not consistent with LOCA conditions 3. Ruptured S/G is also faulted outside of *containment* 4. Primary-to-secondary leakage ≥10 gpm with non-isolable steam release from affected S/G *to the environment* 5. Containment isolation required and containment isolation or ventilation valve(s) not closed when required **AND** Radiological release pathway to the environment exists 6. Inability to isolate any primary system discharging outside containment **AND** Radiological release pathway to the environment exists 7. Emergency Director judgment	1. Conditions requiring entry into Containment-RED path (CSP-Z.1) 2. Containment pressure ≥60 psig and rising (Containment-RED path, CSP-Z.1) 3. Hydrogen concentration in containment ≥6% 4. Containment pressure ≥25 psig with less than one train of containment spray and two containment accident fan cooler units operating 5. CET readings ≥1200°F (Core Cooling-RED path, CSP-C.1) **AND** Restoration procedures not effective with 15 min. 6. CET >700°F with RVLIS NR <25 ft and no RCPs *running* (Core Cooling-RED path, *(CSP-C.1)* **AND** Restoration procedures not effective within 15 min. 7. Containment rad monitor reading ≥15,900 R/hr 8. Emergency Director judgment

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SE	Alert	UE
1(2)RE 212	- - -	- - -	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 305	- - -	- - -	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 307	1.44E1 μCi/cc	1.44E0 μCi/cc	- - -	- - -
1(2)RE 309	1.44E1 μCi/cc	1.44E0 μCi/cc	- - -	- - -
RE 214	- - -	- - -	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 315	- - -	- - -	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 317	2.63E0 μCi/cc	2.63E-1 μCi/cc	- - -	- - -
RE 319	2.63E0 μCi/cc	2.63E-1 μCi/cc	- - -	- - -
1(2)RE 215	- - -	- - -	5.42E+2 μCi/cc	5.42E0 μCi/cc
RE 225	- - -	- - -	2.72E+2 μCi/cc	2.72E0 μCi/cc
RE 226	8.65E+3 μCi/cc	8.65E+2 μCi/cc	- - -	- - -
RE 221	- - -	- - -	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 325	- - -	- - -	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 327	4.34E0 μCi/cc	4.34E-1 μCi/cc	- - -	- - -
RE 224	- - -	- - -	4.18E-1 μCi/cc	4.18E-3 μCi/cc
1(2)RE 231/232	- - -	- - -	- - -	- - -
1(2)SG A/B				
1 ARV	6.69E-1 μCi/cc	6.69E-2 μCi/cc	N/A	N/A
1 SRV	2.48E-1 μCi/cc	2.48E-2 μCi/cc	N/A	N/A
2 SRV	1.24E-1 μCi/cc	1.24E-2 μCi/cc	N/A	N/A
3 SRV	8.25E-2 μCi/cc	8.25E-3 μCi/cc	N/A	N/A
4 SRV	6.20E-2 μCi/cc	6.20E-3 μCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.56E-3 μCi/cc	5.56E-5 μCi/cc
RE 230*	N/A	N/A	7.40E-2 μCi/cc	7.40E-4 μCi/cc

* with Waste Water Effluent discharge not isolated

Table H-1 Safe Shutdown Areas

- 1(2) Containment Building
- Primary Auxiliary Building
- Turbine Building
- Control Building
- Diesel Generator Building
- Gas Turbine Building
- Circ Water Pump House

Table R-2 Dose Proj. / Env. Measurement Class. Thresholds

	GE	SE	Alert
TEDE	1000 mRem	100 mRem	10 mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr. of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

HOT CONDITIONS (RCS > 200°F)

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | D Defueled

NMC

PBNP Emergency Action Level Matrix
EPIP 1.2, Emergency Classification

Approved: ______________ Manager Emergency Preparedness ______________ Date

COLD CONDITIONS

		GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
Loss of AC Power Sources		NONE	NONE	MA8.2 5 6 D Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥ 15 min.	MU8.1 5 6 D Unplanned loss of offsite AC power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥ 15 min.
Loss of DC Power Sources		NONE	NONE	NONE	MU9.1 5 6 ≤105 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for ≥ 15 min. due to unplanned activities
RCS	Temperature	NONE	NONE	MA13.1 5 6 An unplanned event results in RCS temperature exceeding 200°F for ≥ Table M-1 duration	MU13.1 5 6 An unplanned event results in RCS temperature ≥200°F **OR** Loss of all RCS temperature and Reactor Vessel level indication for ≥ 15 min.
RCS	Pressure	NONE	NONE	MA14.1 5 6 Unplanned RCS pressure rise ≥ 10 psig due to loss of decay heat removal	NONE
RCS	Level	MG15.1 5 6 1. Core uncovery for ≥ 30 min. as indicated by <u>EITHER</u> of the following: RVLIS NR ≤ [30 ft] 27 ft **OR** One or more of the following when Reactor Vessel water level cannot be monitored: - Containment High Range Radiation Monitor reading ≥ 10 R/hr - Erratic Source Range Monitor indication - Unexplained sump level rise **AND** 2. Containment challenged as indicated by one or more of the following: - Containment closre not established - Hydrogen concentration in containment 6% - Containment pressure ≥ 60 psig	MS15.1 5 6 With containment closure <u>not</u> established, RVLIS NR ≤ [33 ft] 30 ft **OR** With containment closure established, RVLIS NR ≤ [30 ft] 27 ft MS15.2 5 Reactor Vessel level cannot be monitored for ≥ 30 min. **AND** A loss of Reactor Vessel inventory as indicated by <u>EITHER</u>: Unexplained Containment Sump A level rise **OR** Erratic Source Range Monitor indication MS15.3 6 Reactor Vessel level cannot be monitored **AND** Indication of core uncovery as evidenced by one or more of the following: - Containment High Range Radiation Monitor reading ≥ 10 R/hr - Erratic Source Range Monitor indication - Unexplained Containment Sump A level increase	MA15.1 5 6 Loss of RCS or Reactor Vessel inventory as indicated by <u>EITHER</u>: LI-447 and LI-447A ≤ 0% when aligned **OR** If RCS or Reactor Vessel level cannot be monitored for ≥ 15 min., loss of inventory as indicated by unexplained Containment Sump A level rise	MU15.1 6 Unplanned RCS level lowering below 77.1% (1 foot below RPV flange) for ≥ 15min. **OR** If Reactor Vessel level cannot be monitored, loss of RPV inventory as indicated by unexplained Containment Sump A level rise

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SE	Alert	UE
1(2)RE 212	---	---	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 305	---	---	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 307	1.44E1 μCi/cc	1.44E0 μCi/cc	---	---
1(2)RE 309	1.44E1 μCi/cc	1.44E0 μCi/cc	---	---
RE 214	---	---	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 315	---	---	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 317	2.63E0 μCi/cc	2.63E-1 μCi/cc	---	---
RE 319	2.63E0 μCi/cc	2.63E-1 μCi/cc	---	---
1(2)RE 215	---	---	5.42E+2 μCi/cc	5.42E0 μCi/cc
RE 225	---	---	2.72E+2 μCi/cc	2.72E0 μCi/cc
RE 226	8.65E+3 μCi/cc	8.65E+2 μCi/cc	---	---
RE 221	---	---	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 325	---	---	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 327	4.34E0 μCi/cc	4.34E-1 μCi/cc	---	---
RE 224	---	---	4.18E-1 μCi/cc	4.18E-3 μCi/cc
1(2)RE 231/232	---	---	---	---
1(2)SG A/B				
1 ARV	6.69E-1 μCi/cc	6.69E-2 μCi/cc	N/A	N/A
1 SRV	2.48E-1 μCi/cc	2.48E-2 μCi/cc	N/A	N/A
2 SRV	1.24E-1 μCi/cc	1.24E-2 μCi/cc	N/A	N/A
3 SRV	8.25E-2 μCi/cc	8.25E-3 μCi/cc	N/A	N/A
4 SRV	6.20E-2 μCi/cc	6.20E-3 μCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.56E-3 μCi/cc	5.56E-5 μCi/cc
RE 230*	N/A	N/A	7.40E-2 μCi/cc	7.40E-4 μCi/cc

* with Waste Water Effluent discharge not isolated

Table R-2 Dose Proj. / Env. Measurement Class. Thresholds

	GE	SE	Alert
TEDE	1000 mRem	100 mRem	10 mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr. of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

Table M-1 RCS Reheat Duration Thresholds

Containment and RCS Barrier Status	Duration (min.)
1. RCS intact	60*
2. Containment closure established **AND EITHER**: RCS not intact **OR** RCS reduced inventory	20*
3. Containment closure <u>not</u> established **AND** RCS <u>not</u> intact	0

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is <u>not</u> applicable.

Table H-1 Safe Shutdown Areas

- 1(2) Containment Building
- Primary Auxiliary Building
- Turbine Building
- Control Building
- Diesel Generator Building
- Gas Turbine Building
- Circ Water Pump House

COLD CONDITIONS (RCS ≤ 200°F)

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | D 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** Unit **PB0** Usage Level **Reference** Proposed Rev No ~~43~~ 44 (ew 6/15/04)

Title **Emergency Classification** Classification **NNSR**

☒ Revision ☐ Cancellation ☐ New Document ☐ Other (e.g., periodic review, admin hold) ____

List Temporary Changes/Feedbacks Incorporated: ____

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

Training review recommended Per NP 1.1.3? ☐ NO ☒ YES (If Yes, RFT Number Per NP 1.13 CA 052692 ~~2~~ 3 MS 4/20/04)

Document Preparer (print/sign) Pat Schwartz / [signature] Date 04/15/2004

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 C. JOHNSON / [signature] Date 5/3/04

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) RD 5/3/04 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 / [signature] Date 5/3/04

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) Tim Shaw / [signature] Date 5/12/04

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.

[signature] 5/19/04 [signature] 6/15/04

PORC Meeting No. 2004-036/2004-039

Approval Authority (print/sign) MONICA RAY / [signature] Date 6/16/04

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): ____

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL CONTINUATION

Page ___ of ___

Doc Number **EPIP 1.2** Revision ~~43~~ 44 (MD) 6/15/04 Unit **PBO**

Title **Emergency Classification**

Temporary Change Number ______

Description of Changes:

Step *	**Change/Reason** 5 MS 5/3/04
Total Rewrite	Changed from NUREG 06~~9~~54 scheme to NEI 99-01 scheme.
3.3	Deleted step – references NUREG 0654 Appendix 1
3.5	Changed from: continuously reference both, to: Monitor and deleted "in this procedure" due to EALs now in newly created procedure EPIP 1.2.1, Emergency Action Levels Technical Basis.
Note prior to existing step5.1	Removed as new charts are smaller and in EPIP 1.2 Attachment A
5.1.1	Delete. Following note directs to Fission Product Barriers in EPIP1.2.1~~, Attachment 2~~ MS 5/3/04
Note after existing step 5.1.1	Remove Category 1 and parentheses around Fission Product Barriers. Reason: To meet new EAL scheme and procedure.
Note after existing step 5.1.1	Change From: Attachment C To: EPIP 1.2.1 ~~Attachment 2~~ MS 5/3/04 Reason: To meet new EAL scheme procedure.
5.1.2	Change: From: Attachment C To: Attachment A Reason: New Chart in Attachment A
5.1.4	Change attachment B to EPIP 1.2.1~~, Attachment 1 or 2~~ Reason: New EAL scheme is mode dependent and has Hot and Cold considerations. MS 5/3/04
5.1.5	Delete Reason: EALs are reviewed in full per previous step to determine the EAL that applies.
Note prior to 5.1.6.c	Delete Reason: Duplicates instructions in step 5.1.6.c.
5.1.6.c	Delete from "IF to THEN". Capitalize R in Return. Reason: Provides direction for EPIP 1.1 usage no matter where EPIP 1.2 is being used.
Reference Section	Add: 6.20 Regulatory Guide 1.101, Rev. 4 – Emergency Planning and Preparedness For Nuclear Power Reactors Reason: Acceptance of NEI 99-01 as an Alternative Methodology for the Development of Emergency Action Levels.
Bases	Delete B-2 NUREG-0654/FEMA-REP 1 basis Reason: New Basis is NEI 99-01
Bases	Change B-3 to B-2 and change to read: NEI 99-01, Rev 4 / NUMARC NESP-007, Methodology for Development of Emergency Action Levels. Reason: New scheme of EALs.
B-4	Delete Reason: References NUREG 0654 – old basis.
Note after B-4	Delete Reason: References NUREG 0654 – old basis.
Attachment A	Replace with new Attachment A ~~– same as EPIP 1.2.1 Attachment 1 and 2. Reason: New scheme EALs.~~ MS 5/3/04
Pages 10 – 97	Delete Reason: Old scheme and replaced by new procedure EPIP 1.2.1

Other Comments

* Note: Recording of Step Number(s) is not required for multiple occurrences of identical information or when not beneficial to reviewers.

Point Beach Nuclear Plant

PROCEDURE PREPARER/TECHNICAL REVIEWER CHECKLIST

PART I- Preparer Checklist (page 1 of 2)

Procedure Number **EPIP 1.2** Revision ~~43~~ 44 CMB 6/15/04 Unit **PB0**

Title **Emergency Classification**

	Review Requirements	YES	N/A
1.	Ensure the procedure purpose is clearly stated and the procedure accomplishes the purpose.	☒	☐
2.	Ensure necessary precautions, prerequisites and controls are included to address potential industrial safety hazards and radiation safety hazards such as personnel protection, hazardous materials, waste or environments. Consult Industrial Health and Safety or Radiation Protection as necessary.	☐	☒
3.	Ensure responsibilities and qualifications for procedure performance are clearly defined (e.g., licensed SRO, NDE Level III, VT-2 examiner, certified QC inspector, position responsible for making acceptance criteria determinations, etc.).	☒	☐
4.	Ensure the procedure complies with applicable writer's guide (PBNP Site guide or AOP/EOP guide) for format and content requirements and is written in a manner that is easily followed and understandable.	☒	☐
5.	Ensure action steps are written as short and concise sentences and the number of actions in each step is limited to one, unless the actions are functionally related, AND the actions can be performed at the same time.	☒	☐
6.	Ensure steps return equipment to original or desired condition, adequately address post maintenance testing requirements and, specify any required follow-up actions to maintain status control, appropriate plant configuration, etc.	☐	☒
7.	Ensure references identified in the procedure are correct (e.g. correct Unit, train, set points, etc. Procedures and other documents being referenced are correctly identified, not cancelled, etc.).	☒	☐
8.	Ensure the procedure complies with the references, including codes, standards, design bases documents, drawings, vendor manuals and other documents (review applicable documents or data sources as necessary).	☒	☐
9.	Ensure the procedure complies with Current or ISFSI Licensing Basis requirements (as defined in NP 5.1.8 and NP 5.1.7) such as Tech Specs, FSAR, and will not cause any violations or discrepancies such as involuntary Tech Spec Action Conditions, etc. (review applicable documents or data sources as necessary).	☒	☐
10.	Ensure all regulatory commitments and Quality Assurance requirements are incorporated by reviewing the applicable documents or data sources.	☒	☐

PART I- Preparer Checklist (page 2 of 2)

	Review Requirements	YES	N/A
11.	Ensure the procedure addresses applicable design, vendor, scientific and technical information (review information as appropriate).	☒	☐
12.	Ensure the procedure adequately addresses related program requirements [e.g., Reactivity Management (NP 7.6.7), Foreign Materials Exclusion (NP 8.4.10), Infrequently Performed Tests or Evolutions (NP 1.2.6), Temporary Mods (NP 7.3.1, etc.]. (B-1, B-2, B-10). Review documents as necessary.	☒	☐
13.	Ensure appropriate Configuration Management has been addressed (e.g., impact on other equipment, documents, software or simulator. Operator aids identified, equipment names and numbers agree with Plant equipment and other documents or data sources, etc.). Ensure necessary changes are initiated.	☐	☒
14.	Ensure internal and industry operating experience has been addressed in the procedure by reviewing related action requests, other applicable "t-Track" items, INPO, NRC, EPRI documents, etc.	☒	☐
15.	Ensure required forms are completed and attached (e.g., Infrequently Performed Tests or Evolutions, Temporary Mods, CHAMPS callups and equipment record sheets processed, etc.). (B-1, B-2, B-9, B-10)	☐	☒

Comments

Performed By PATRICK J. Schwartz / [signature] Date 5/3/04
(Procedure Writer)

Point Beach Nuclear Plant

PROCEDURE PREPARER/TECHNICAL REVIEWER CHECKLIST

PART II- Technical Reviewer Checklist (page 1 of 2)

Note: The Technical Reviewer may also use or review the questions in Part I to aid review. For revisions not considered total rewrites, consider the checklist questions for the changes being made (at minimum) and the effect of the changes on the procedure.

Procedure Number **EPIP 1.2** Revision ~~43~~ 44 MD 6/15/04 Unit **PB0**

Title **Emergency Classification**

	Review Requirements	YES	NO	N/A
1.	Ensure necessary precautions, prerequisites and controls are included to address potential industrial safety hazards and radiation safety hazards such as personnel protection, hazardous materials, waste or environments. Consult Industrial Health and Safety or Radiation Protection as necessary.	☐	☐	☒
2.	Ensure the procedure contains all limitations and cautions required to protect personnel or prevent equipment damage, including equipment limitations or interference with other equipment requirements such as EQ or Appendix R requirements.	☒	☐	☐
3.	Ensure the procedure is technically correct and the performance methods (flow paths, testing methodology, maintenance methods, operating sequence, etc.) are correct to preclude unexpected operability issues and maintain appropriate plant configuration (perform walk-downs as appropriate and review documents such as drawings, vendor manuals, interfacing procedures, etc.).	☐	☐	☒
4.	Ensure appropriate contingency actions are addressed such as appropriate actions for unacceptable procedure results, acceptance criteria, etc.	☒	☐	☐
5.	For procedure changes that manipulate equipment (valves, breakers, switches, locks, etc.), ensure restoration lineups and independent verification are provided as required, OR procedures are referenced that provide required restoration and independent verification. (B-6, B-7)	☐	☐	☒
6.	Ensure procedure steps are written to avoid preconditioning of equipment (e.g., avoid actions that could influence test results). (B-8)	☐	☐	☒
7.	Ensure steps that require, or may require, entry into a Tech Spec Action Condition are identified (review Tech Specs, etc. as necessary).	☐	☐	☒
8.	Ensure hold point and independent verification processes have been appropriately used in the procedure (reference NP 8.4.1, NP 2.1.2). (B-5)	☐	☐	☒

Point Beach Nuclear Plant

PROCEDURE PREPARER/TECHNICAL REVIEWER CHECKLIST

PART II- Technical Reviewer Checklist (page 2 of 2)

	Review Requirements	YES	NO	N/A
9.	Ensure qualitative and quantitative acceptance criteria values clearly indicate acceptable values and tolerances, the values are correct, and consistent with requirements such as Tech Specs, ASME codes, etc. Review documents as necessary.	☒	☐	☐
10.	Ensure the procedure complies with the references, including codes, standards, design bases documents, drawings, vendor manuals and other documents (review applicable documents or data sources as necessary).	☒	☐	☐
11.	Ensure the procedure complies with Current or ISFSI Licensing Basis requirements (as defined in NP 5.1.8 and NP 5.1.7) such as Tech Specs, FSAR, and will not cause any violations or discrepancies such as involuntary Tech Spec Action Conditions, etc. (review applicable documents or data sources as necessary).	☒	☐	☐

Comments (Explain any "NO" Answers):

Performed By RICHARD C. JOHNSON / Richard C. Johnson Date 5/3/04
(Technical Reviewer)

Point Beach Nuclear Plant

10 CFR 50.59/72.48 APPLICABILITY FORM

Brief Activity Title or Description: **Revise EPIP 1.2 Emergency Classification**

This form is required to be completed and attached to the applicable activity change forms to document all or portions of an activity that are covered by another regulation other than 10 CFR 50.59 and 10 CFR 72.48 (pre-screening criteria 2). See NP 5.1.8, 10 CFR 50.59/72.48 Applicability, Screening and Evaluation (New Rule).

NOTE: Guidance for searching the FSAR, Technical Specifications, Regulatory Commitments (CLB Commitment Database) and other licensing basis documents can be found in NP 5.1.8, Attachment G.

NOTE: Although 10 CFR 50.59 and 72.48 may not be applicable to the processes listed below, change activities conducted under these processes may require changes to the FSAR. If so, initiate FSAR changes per NP 5.2.6, FSAR Revisions.

	Regulatory or Plant Process	YES	NO
1.	Does the activity require a change to the Facility Operating License, License Conditions or Technical Specifications? (If the answer is **YES**, process the applicable changes per NP 5.2.7, License Amendment Request Preparation, Review and Approval.)	☐	☒
2.	**NOTE: The Quality Assurance Plan is described in FSAR Section 1.4.** Does the activity require a change to the Quality Assurance Program? If the answer is **YES**, process the applicable changes per NP 11.1.3, QA Program Revisions.	☐	☒
3.	**NOTE: Implementation of Security Plan changes that require physical changes to the plant, or changes to operator access to the plant require a screening.** **NOTE: Security is described in FSAR Section 12.7.** Does the activity require a change to the PBNP Security Plan, a safeguards contingency plan, or security training and qualification plan? If the answer is **YES**, assess the acceptability of the change per 10 CFR 50.54(p) using Security procedures.	☐	☒
4.	**NOTE: The Emergency Plan is described in FSAR Section 12.6.** Does the activity require a change to the Emergency Plan? If the answer is **YES**, assess the acceptability of the change per 10 CFR 50.54(q) using NP 1.8.3, 10 CFR 50.54(q) Evaluations.	☒	☐
5.	**NOTE: The Radiation Protection Program is described in FSAR Section 11.4.** Does the activity involve a change to the PBNP Radiation Protection Program or its implementing procedures, **AND** the activity in its entirety is within the requirements of 10 CFR 20, Standards for Protection Against Radiation?	☐	☒
6.	**NOTE: Changes to the plant or method of evaluation that result in re-analysis of the FSAR loss-of-coolant accident (LOCA) analysis require a screening.** Does the activity require a change to the FSAR LOCA analysis results subject to 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors? If the answer is **YES**, process the applicable changes per NP 5.2.12, 10 CFR 50.46 Reporting Requirements, and NP 5.2.6, FSAR Revisions.	☐	☒
7.	**NOTE: Regulatory commitments are found in the CLB Commitment Database.** Does the activity involve a change to a Regulatory Commitment ? If the answer is **YES**, process the applicable changes per NP 5.1.7, Regulatory Commitment Changes.	☐	☒

10 CFR 50.59/72.48 APPLICABILITY FORM

	Regulatory or Plant Process	YES	NO
8.	Does the activity involve a change to the Environmental Manual (EM), Radiological Effluent Control Program Manual (RECM), Offsite Dose Calculation Manual (ODCM), or Process Control Program (PCP), <u>AND</u> does <u>NOT</u> involve changes in use of explosive gases in waste treatment systems? If the answer is <u>YES</u>, document the applicable changes per the requirements of TS 5.5.1.	☐	☒
	NOTE: For purposes of determining 10 CFR 50.59 / 72.48 applicability, the determination of an administrative procedure below takes precedence over definitions or classifications in other plant procedures or guidelines.		
9.	Does the activity involve initial issue or a change to an administrative procedure or controlled document, <u>OR</u> an administrative software change <u>ONLY</u>? <u>ALL</u> of the following statements have to be marked YES for the procedure, controlled document, or software change to be considered administrative.	☒	☐
	a. **<u>DOES NOT</u>** direct, control or provide data as to how plant structures, systems, or components are operated, maintained, tested or repaired either specifically **<u>OR</u>** generically.	☒	☐
	b. **<u>DOES NOT</u>** specify acceptance criteria or operating limits for plant structures, systems, or components or demonstrates criteria/limits are met.	☒	☐
	c. **<u>DOES NOT</u>** specify parts, materials, chemicals, lubricants, etc. to be used in plant structures, systems, or components.	☒	☐
	d. **<u>DOES NOT</u>** specify compensatory action(s) to address plant structures, systems, or components out of service, or to address non-conforming conditions.	☒	☐
	e. **<u>DOES NOT</u>** affect operator access to operating areas of the plant.	☒	☐

10 CFR 50.59/72.48 APPLICABILITY CONCLUSION

NOTE: If <u>ANY</u> portion of the activity is <u>NOT</u> controlled by one or more of the processes above, further 10 CFR 50.59 / 72.48 review is required (i.e., portions not covered by the above processes shall be prescreened to other criteria or screened).

	YES	NO
<u>ALL</u> aspects of the activity are controlled by one or more of the processes above, therefore <u>NO</u> additional 10 CFR 50.59 and 72.48 review is required.	☒	☐

If the above question is answered <u>NO</u>, briefly describe the portions of the activity <u>NOT</u> covered by one or more of the above processes:

N/A

	Name (Print) / Signature		Date
Performed By	PATRICK J. SCHWARTZ / Patrick J Schwartz	Date	5/3/04
Reviewed By	RICHARD C. JOHNSON / Richard C Johnson	Date	5/3/04

Point Beach Nuclear Plant

10 CFR 50.54(q) EVALUATION CHECKLIST

Document EPIP 1.2 Title Emergency Classification Rev ~~43~~ 44 MR 6/15/04

1. Describe change:

Revise to address EAL scheme change to NEI 99-01.

2. Indicate which of the following standards of 10 CFR 50.47(b) may be affected by the change.

A. Assignment of Responsibility	☐
B. Onsite Emergency Organization	☐
C. Emergency Response Support and Resources	☐
D. Emergency Classification System	☒
E. Notification Methods and Procedures	☐
F. Emergency Communications	☐
G. Public Education and Information	☐
H. Emergency Facilities and Equipment	☐
I. Accident Assessment	☒
J. Protective Response	☐
K. Radiological Exposure Control	☐
L. Medical and Public Health Support	☐
M. Recovery and Reentry Planning and Post-accident Operations	☐
N. Exercises and Drills	☐
O. Radiological Emergency Response Training	☐
P. Responsibilities for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plan	☐

Does the change/revision result in the loss of ability to meet any of the standards of 10 CFR 50.47(b) or any NRC approved alternatives to those standards? (Specifically discuss any item checked above, explaining why the standard is, or is not, being met). ☐ YES ☒ NO

D. Emergency Classification System - Convert from NUREG 0654 to NEI 99-01 Rev 4.

I. Accident Assessment - Considers mode applicability in the new scheme.

3. Indicate which of the following areas of 10 CFR 50, Appendix E may be affected by the change.	
(i)(ii)(iii) Emergency plan as described in the FSAR	☐
(iv) A. Organization for coping with radiological emergencies	☐
(iv) B. Assessment of radiological emergencies	☐
(iv) C. Classifications, EALs and ERO Activation	☒
(iv) D. Notification of Federal, State and local agencies and the public	☐
(iv) E. ERFs, equipment, and communications	☐
(iv) F. Training, drills, and exercises	☐
(iv) G. Plans and procedures and surveillance of equipment and supplies	☐
(iv) H. Re-entry and Recovery following an accident	☐
(v) Emergency Response Data System (ERDS)	☐

Does the change/revision result in the loss of ability to meet any of the requirements of 10 CFR 50, Appendix E or any NRC approved alternatives to those requirements? ☐ YES ☒ NO
(Specifically discuss any item checked above, explaining why the requirement is, or is not, being met).

(IV) C. New scheme - leaving NUREG 0654, going to NEI 99-01 Rev 4.

4. Does the change/revision result in a reduction of any commitment that is not justified by the basis for that commitment? ☐ YES ☒ NO

Justification for Answer:

RG 1.101 Rev 4 accepts the new scheme of EALs (99.01 Rev 4)

5. Does the change/revision decrease the effectiveness of the emergency plan (including EALs)? ☒ YES ☐ NO

Justification for Answer:
(Explain how the change / revision will not reduce the capability or resources for emergency response or explain why the change/revision will result in a commensurate reduction in the need for such capabilities and resources.)

New scheme addresses mode applicability and may result in different classification.

6. 50.54(q) Evaluation

______ Proposed change DOES NOT require prior NRC approval (all questions above answered "No")
___X___ Proposed change DOES require prior NRC approval (any question above answered "Yes")

Prepared by: PATRICK J. SCHWARTZ / Patrick J. Schwartz Date: 5/3/04
Reviewed by: James W. Connolly / James W. Connolly Date: 05/03/04
Regulatory Affairs
Approved by: MONICA RAY / Monica Ray Date: 5/3/04
EP Manager

WORD PROCESSING DOCUMENT CHECKLIST

Routing	Extension	Date	Action Or Comments

TEMPORARY CHANGE TO BE MADE PERMANENT? ☐ YES TEMP CHANGE NO. ____________

DOC. ID: EDIP 1.2 NEW REV: ________

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INITIAL PROCESSING **DATE IN:**	4/15
Update **Master Index** with your name and special comments if necessary.	✓
Verify **TITLE, USAGE LEVEL, CLASSIFICATION** and **CURRENT REVISION** against master for *uncontrolled* procedures.	✓
Connect to Macros and run set-up macros: ▪ Options (shift alt ctrl O) ▪ TNR 12 (shift alt ctrl N) ▪ Non-breaking Hyphens (shift alt ctrl -) ▪ Remove all Rev Bars (shift alt ctrl R)	✓
Update Header/Footer: ▪ Up the Revision # and type DRAFT ▪ Usage Level ▪ Current Date ▪ Check classification in header with cover page and PBF-0026a/c. No classification for controlled reference documents and administrative procedures.	✓
Change MSS (Manager's Supervisory Staff) to PORC (Plant Operation's Review Committee)	✓
Temp Changes: Ensure all changes are incorporated by verifying them against PBF-0026c. (DO NOT change Approval Authority/Reviewer on Permanent Temp Changes.)	✓
Make appropriate changes and rev bar. (Rev 0's and Total Rewrites **DO NOT** get rev barred)	✓
Check bookmarks/cross-references.	✓
Ensure reference titles are correct and still effective. **(Not Including Temp Changes)**	✓
Rerun Non-breaking hyphen macro. Spell check document and turn hidden text off before printing.	✓
Print and proofread document: ▪ Page endings and step numbering are correct (including Table of Contents) ▪ Error References ▪ Page Numbers ▪ Attachments are included	✓
Save copy in EDMS, and in EDMS Backup. List here, if different.____________	✓
Delete electronic copy in Draft Transfers after saving in EDMS Backup.	✓
For first time clean-ups: Attach yellow Conversion Verification Form.	Yes ✓ ~~No~~
Update Master Index with date forwarded to originator/Distribution.	✓

SUBSEQUENT REVISIONS **DATE IN:**					
Make appropriate changes and rev bar.					
Rerun Non-breaking hyphen macro. Spell check document and turn hidden text off before printing.					
Remove hidden text, print and proofread document.					
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Update Master Index.					
DATE OUT:	4/15				
INITIALS:	[illegible]				

Date Returned for Issue:

Point Beach Nuclear Plant
PROCEDURE VALIDATION

Procedure Number **EPIP 1.2** Revision ____ Unit **PB0**

Title **Emergency Classification**

METHOD OF VALIDATION

(Methods may be combined when necessary)

☐ Walk-down (normally used unless plant conditions prohibit or ALARA concerns are involved)

☒ Simulator ☐ Simulation ☒ User group review ☒ Other **Operations Support Services, Inc.**

	REVIEW REQUIREMENTS	YES	NO	N/A
1.	Ensure the procedure accomplishes the stated purpose without introducing new challenges to personnel or equipment (e.g., no reactivity issues or involuntary entry to Current Tech Spec LCO or Improved Tech Spec Action Condition. Proper flow paths, maintenance, or testing methods identified).	☒	☐	☐
2.	Ensure the procedure contains all required processes and steps and can be performed as written and in sequence written.	☒	☐	☐
3.	Ensure the procedure is written such that qualified users should be able to perform it consistently.	☒	☐	☐
4.	Ensure Industrial and Radiological Safety have been appropriately addressed for personnel protection.	☒	☐	☐
5.	Ensure the procedure effectively addresses the communication and coordination necessary for procedure performance, both internal and external to responsible organization.	☒	☐	☐

Explain "NO" Answers:
(Use additional sheets as necessary) (Continued) ☐

N/A

Comments and Suggestions for Improvement:
(Use additional sheets as necessary) (Continued) ☐

The validation process ensured that the Emergency Action Levels (EALs) were usable and operationally correct. The validation process also identified the ability of validation team members to arrive at consistent interpretations of EALs under varying conditions. The validation team was led by Emergency Planning, supported by Operations Support Services, Inc., and included various site disciplines from operations, engineering, emergency planning, security and operations training. The test methodology was comprised of table-top and control room simulator scenarios requiring the validation team to conduct step-by-step user actions to implement classification of selected EALs. Validation team participants are listed on the attached PBNP Training Attendance Report.

Performed By: **Monica Ray** / Monica Ray Date 5/10/04
(Validator)

Point Beach Nuclear Plant
TRAINING ATTENDANCE REPORT

Activity Code:	Lesson Title	Rev	Initial/ Cont	Date(s)	Hrs	Evaluation ID
1.	OSSI - TABLETOP DRAFT EAL/Simulator use	NA		12/17/03	8.0	
2.	for EAL classification capability.					
3.						
4.						

Trainee's Name *Please print name legibly*	Trainee's Signature	Trainee's ID WE = WE Employee ID # WPS Union = WPS ID # Contractor = Social Sec. # C# = NMC Train ID #	Activity Status / Grade 1 Status	1 Grade	2 Status	2 Grade	3 Status	3 Grade	4 Status	4 Grade
1. PATRICK J. Schwartz	Patrick J Schwartz	XX7215								
2. James E Schleif	James E Schleif	WE4809								
3. Steve Boone	S Boone	PB4807								
4. Charles Stauzer	[illegible]	W33913								
5. Jim Green	J Green	WE3333								
6. Tim Lowe	Timothy [illegible]	WE5462								
7. Tim TAylor	[illegible]	WE6307								
8. Rick MERKES	[illegible]	PB1539								
9. Daniel C. Pond	Daniel C Pond	PB3143								
10.										

Instructor's Name *(Please print)*	Instructor's Signature	SME*	Instructor's ID	1		2		3		4	
KELLY WALKER	[illegible]										
Edward G. Bates	E G Bates										

Comments: M L DAUS M L Davis

PBF-6401
Revision 10 07/31/03
File: A8.16

FRONT

Status values:
- P (Successfully completed activity requirements)
- F (Failed activity requirements)
- A (Attended without evaluation)
- I (lesson incomplete – makeup required)
- T (Taught – Registrar entry desired) *use "X" if Reg. entry not wanted*
- X (No registrar entry desired – e.g., trainee did not attend activity)

Grade values: % (e.g., 95.2)

* Checkmark SME column if instructor is an SME (versus a Training instructor)

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

____________ ____________
____________ ____________
____________ ____________

TABLE OF CONTENTS

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 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

TIME / INITIALS

NOTE: If the EAL relates to Fission Product Barriers, EPIP 1.2.1 provides additional information on the CHALLENGE and LOSS criteria.

5.1.1 Identify the status of Fission Product Barriers from Attachment A, as required.

	Intact	Challenge	Loss
Fuel Clad			
RCS			
Containment			

/

5.1.2 Make an initial EAL selection from Attachment A.

/

NOTE: A challenge to, or loss of, a barrier should not be anticipated unless the trend is rapid, and the values are close to the threshold/criteria.

5.1.3 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.

/

TIME / INITIALS

NOTE: Classifications are to be made consistent within 15 minutes once plant parameters reach an Emergency Action Level (EAL) indication in the Control Room.

5.1.4 **IF** an event has been categorized and the threshold of the EAL and related conditions are verified to have been met or exceeded
THEN declare the emergency. ____ / ____

a. Record the time of declaration, the emergency classification, and the EAL

Classification __________ EAL ___________ ____ / ____

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.

TIME / INITIALS

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 (per NUREG-0654, II.A.1.d, II. B.2).

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.

/

6.0 REFERENCES

6.1 Point Beach 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

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 EMERGENCY	ALERT	UNUSUAL EVENT
Reactor Fuel	Inadvertent Criticality	NONE	NONE	NONE	**MU1.1** An unplanned sustained positive startup rate observed on nuclear instrumentation
Reactor Fuel	Coolant Activity	NONE	NONE	NONE	**MU2.1** Coolant activity ≥0.8 µCi/gm dose equivalent I-131
Reactor Fuel	Failed Fuel Monitor	NONE	NONE	NONE	**MU3.1** Failed Fuel Monitor (RE-109) ≥24 mrem/hr not due to a planned evolution
Reactor Fuel	Refueling Accidents & Other Radiation Monitors	NONE	NONE	**MA4.1** Confirmed sustained (10 min average) alarm on any of the following radiation monitors resulting from an uncontrolled fuel handling process indicating damaged or uncovered irradiated fuel: - RE-105 SFP Area Low Range Area Radiation High Alarm (≥10 mR/hr) - RE-135 SFP Area High Range Area Radiation High Alarm (≥100 mR/hr) - 1(2) RE-211 Containment Air Particulate Monitor High Alarm (≥0.5 µCi) - 1(2) RE-212B Containment Background Monitor High Alarm (≥100 mR/hr) **MA4.2** Report of visual observation of irradiated fuel uncovered **OR** Loss of refueling water inventory as indicated by excessive makeup rate or unexpected lowering in refueling water storage tank level	**MU4.1** Spent fuel pool (reactor cavity during refueling) water level cannot be restored and maintained above the spent fuel pool water level alarm setpoint **AND** Unplanned SFP Area Radiation Monitor readings rise - RE-105 SFP Area Low Range Area Radiation Monitor - RE-135 SFP Area High Range Area Radiation Monitor
RCS	Leakage	NONE	NONE	NONE	**MU5.1** Unidentified or pressure boundary leakage ≥ 10 gpm **OR** Identified leakage ≥ 25 gpm
Comm. Loss		NONE	NONE	NONE	**MU6.1** Loss of all communications capability affecting the ability to **EITHER** Perform routine operations **OR** *Notify offsite agencies or personnel*
Radioactivity Release/Area Radiation	Effluent Monitors	**RG1.1** A valid reading on any monitors listed in Table R-1 column "GE" for ≥15 min. unless dose assessment can confirm releases are below Table R-2 column "GE" within this time period	**RS1.1** A valid reading on any monitors listed in Table R-1 column "SE" for ≥15 min. unless dose assessment can confirm releases are below Table R-2 column "SE" within this time period	**RA1.1** Loss of control of radiatioactive materials as indicated by a valid reading on any monitors listed in Table R-1 column "ALERT" for ≥15 min. unless sample analysis can confirm release rates ≤200 x ODCM limits within this time period	**RU1.1** Loss of control of radiatioactive materials as indicated by a valid reading on any monitors listed in Table R-1 column "UE" for ≥60 min. unless sample analysis can confirm release rates ≤2 x ODCM limits within this time period
Radioactivity Release/Area Radiation	Dose Proj./Env. Meas./Release Rates	**RG2.1** Dose projections or field surveys resulting from actual or imminent release which indicate doses / closed - window dose rates ≥Table R-2 column "GE" at the site boundary or beyond.	**RS2.1** Dose projections or field surveys resulting from actual or imminent release which indicate doses / dose rates ≥Table R-2 column "SE" at the site boundary or beyond.	**RA2.1** An unplanned gaseous or liquid release with rates ≥200 x ODCM limits for ≥15 min **RA2.2** Dose projections or field surveys resulting from an unplanned actual or imminent release which indicate doses / dose rates ≥Table R-2 column "*Alert*" *at the site* boundary or beyond.	**RU2.1** An unplanned gaseous or liquid release with rates ≥2 x ODCM limits for ≥60 min.
Radioactivity Release/Area Radiation	Area Radiation Levels	NONE	NONE	**RA3.1** Sustained (10 min. average) area radiation levels ≥15 mR/hr in **EITHER** Control Room (RE 101) **OR** Central Alarm Station (by survey) **OR** Secondary Alarm Station (by survey) **RA3.2** Sustained (10 min. average) abnormal area radiation levels ≥12 R/hr in any Table H-1 Safe Shutdown Area **AND** Access to affected area is required for safe operation or shutdown.	**RU3.1** Any sustained (10 min. average) direct ARM readings ≥100 x alarm or offscale high not resulting from a planned event or evolution
Hazards	Security Threats	**HG1.1** An adversary has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions: - Reactivity control - RCS inventory - Secondary heat removal - Spent Fuel Pool integrity	**HS1.1** Intrusion into a Vital Area by an adversary indicated by notification by the Security Shift Supervisor to implement AOP-29 for Vital Area intrusion.	**HA1.1** Intrusion into the site Protected Area by an adversary indicated by notification by the Security Shift Supervisor to implement AOP-29 for a PA intrusion.	**HU1.1** Indication of attempted sabotage, hostage/extortion, civil disturbance or strike action onsite **OR** Notification of any credible site-specific threat by the Security Shift Supervisor or outside agency (NRC, military or law enforcement)
Hazards	Fire or Explosion	NONE	NONE	**HA2.1** Fire or explosion in any Table H-1 Safe Shutdown Area, which results in **EITHER**: Visible damage to plant equipment or structures needed for safe shutdown **OR** Affected safety system performance is degraded indicating damage to a safety system	**HU2.1** Confirmed fire in the Protected Area not extinguished in ≤15 min. of Control Room Notification **HU2.2** Report by plant personnel of an explosion within Protected Area resulting in visible damage to permanent structures or equipment.
Hazards	Vehicle Crash/Toxic and Flammable Gas	NONE	NONE	**HA3.1** Vehicle crash which precludes personnel access to or damages equipment in one or more Table H-1 Safe Shutdown Areas **HA3.2** Report or detection of toxic or flammable gases within any Table H-1 Safe Shutdown Area in concentrations that **EITHER**: Will be immediately life threatening to plant personnel (IDLH) **OR** Exceed the lower flammability limit	**HU3.1** Vehicle crash into plant structures or systems within the Protected Area **HU3.2** Report or detection of toxic or flammable gases that could enter or have entered within the Protected Area in amounts that could affect the health of plant personnel or safe plant operation **OR** Report by local, county or state officials for evacuation or sheltering of site personnel based on an offsite event
Hazards	Natural Events	NONE	NONE	**HA4.1** Two or more seismic monitors (SEI 6210 through 6213) indicate ground acceleration **EITHER**: ≥0.06 g horizontal **OR** ≥0.04 g vertical **HA4.2** Sustained (15 min. average) winds > 108 mph onsite **OR** Tornado strikes any Table H-1 Safe Shutdown Area **HA4.3** Uncontrolled flooding that results in degraded safety system performance or that creates industrial safety hazards that precludes access necessary to operate or monitor safety euipment in **EITHER**: - The auxilliary building caused by rupture of the SW header **OR** - The water intake structure caused by rupture of a circulating water system expansion joint or fire water main	**HU4.1** Activation of 2 or more seismic monitors (SEI 6210 through 6213) **AND** Verified by: - Actual ground shaking **OR** - By contacting the U.S. Geological Survey National Earthquake Information Center **HU4.2** Sustained (15 min. average) winds > 75 mph onsite **OR** Report by plant personnel of tornado striking within plant Protected Area **HU4.3** Uncontrolled flooding in the auxiliary building *caused by rupture of the SW header* **OR** Uncontrolled flooding in the water intake structure caused by rupture of a circulating water system expansion joint or fire water main
Control Rm Evacuation		NONE	**HS5.1** Control Room evacuation **AND** Transfer of reactivity, RCS inventory and secondary heat removal control functions cannot be established per AOP-10A Safe Shutdown - Local Control in ≤15 min.	**HA5.1** Entry into AOP-10 Control Room Inaccessibility due to Control Room Evacuation	NONE
Other		**HG6.1** Any event in the judgement of the Emergency Director is in progress which indicates actual or imminent core damage and the potential for a large release of radioactive material in excess of EPA Protective Action Guides outside the site boundary	**HS6.1** Any event in the judgement of the Emergency Director is in progress which indicates actual or likely failures of plant systems needed to protect the public. Any releases are not expected to result in exposures which exceed EPA Protective Action Guides beyond the site boundary	**HA6.1** Any event in the judgement of the Emergency Director, that could cause or has caused actual substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of EPA Protective Action Guides	**HU6.1** Any event, in the judgement of the Emergency Director, that could lead to or has led to a potential degradation of the level of safety of the plant.
ISFSI Events	Loss of Cask Confinement	NONE	NONE	NONE	**IU1.1** Loss of cask confinement boundary as indicated by exceeding any of the following external surface dose rates on any loaded Dry Storage Cask: - ≥ 100 mR/hr at the cask side - ≥ 200 mR/hr at the top of the cask - ≥ 350 mR/hr at the cask air inlet - *≥ 100 mR/hr at the cask air outlet*
ISFSI Events	Security	NONE	NONE	NONE	**IU1.2** Report by Security Shift Supervisor of a security concern within the ISFSI.

HOT CONDITIONS

	GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
ATWS	MG7.1 1 2 Conditions requiring entry into Subcritically-RED path (CSP-S.1) currently exist AND Conditions requiring entry into EITHER: Core Cooling-RED path (CSP-C.1) OR Heat Sink-RED path (CSP-H.1)	MS7.1 1 2 Conditions requiring entry into Critical Safety Function-Subcritically-RED path (CSP-S.1)	MA7.1 1 2 Any failure of the Reactor Protection System to generate an automatic trip signal and reduce power range to ≤5% AND Manual trip successful	NONE
Loss of AC Power Sources	MG8.1 1 2 3 4 Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 AND EITHER: Power restoration to any safety-related 4160 VAC bus or 480 VAC bus is not likely in ≤4 hours OR Conditions require entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)	MS8.1 1 2 3 4 Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥15 min.	MA8.1 1 2 3 4 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 ≥15 min. (one source away from station blackout): - A single emergency diesel generator (G01, G02, G03 & G04) - LVSAT 1(2)-X04 - UAT 1(2)-X02 - Cross-tying with the opposite unit power supply	MU8.1 1 2 3 4 Unplanned loss of offsite AC power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥15 min.
Loss of DC Power Sources	NONE	MS9.1 1 2 3 4 ≤105 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for ≥15 min. due to unplanned activities	NONE	NONE
H4 Equipment Failures — Tech Spec.	NONE	NONE	NONE	MU10.1 1 2 3 4 Plant is not brought to required operating mode within Technical Specifications LCO required action completion time
H4 Equipment Failures — Turbine Failure	NONE	NONE	MA11.1 1 2 3 4 Turbine failure generated missiles resulting in visible damage to or penetrating any Table H-1 Safe Shutdown Area structure or system	MU11.1 1 2 3 4 Report of main turbine failure requiring turbine trip resulting in: Damage to turbine-generator seals OR Casing penetration
H4 Equipment Failures — Loss of Indicators/Alarms	NONE	MS12.1 1 2 3 4 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 min. AND PPCS is unavailable AND Complete loss of ability to monitor all critical safety functions status AND A significant transient is in progress	MA12.1 1 2 3 4 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 min. AND EITHER: A significant transient is in progress OR PPCS is unavailable	MU12.1 1 2 3 4 Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 min.
Fission Product Barriers	FG1.1 1 2 3 4 Loss of any two barriers with a loss or challenge of a third (Table F-1)	FS1.1 1 2 3 4 Loss or challenge of any two barriers (Table F-1)	FA1.1 1 2 3 4 Any loss or challenge of Fuel Cladding or RCS (Table F-1)	FU1.1 1 2 3 4 Any loss or challenge of Containment (Table F-1)

Table F-1 FISSION PRODUCT BARRIER MATRIX

Fuel Clad Barrier		RCS Barrier		Primary Containment Barrier	
☐ Loss	☐ Challenge	☐ Loss	☐ Challenge	☐ Loss	☐ Challenge
1. Conditions requiring entry into Core Cooling-RED path (CSP-C.1) 2. Coolant activity ≥300 uCi/gm I-131 equivalent 3. CET readings ≥1200°F 4. Containment rad monitor reading ≥17 R/hr 5. Failed Fuel Monitor (RE-109) reading ≥120 mRem/hr 6. Emergency Director judgment	1. Conditions requiring entry into Core Cooling-ORANGE path (CSP-C.2) 2. Conditions requiring entry into Heat Sink-RED path (CSP-H.1) 3. CET readings ≥700°F 4. RVLIS NR ≤25 ft with no RCPs running 5. Emergency Director judgment	1. RCS subcooling based on core exit thermocouples ≤[80°F] 35°F due to RCS leakage 2. SGTR in excess of available charging pumps 3. Containment rad monitor reading ≥3.0 R/hr 4. Emergency Director judgment	1. Conditions requiring entry into RCS Integrity-RED path (CSP-P.1) 2. Conditions requiring entry into Heat Sink-RED path (CSP-H.1) 3. Unisolable leak exceeding 60gpm 4. Emergency Director judgment	1. Rapid unexplained containment pressure drop following initial rise 2. Containment pressure or sump level response not consistent with LOCA conditions 3. Ruptured S/G is also faulted outside of containment 4. Primary-to-secondary leakage ≥10 gpm with non-isolable steam release from affected S/G to the environment 5. Containment isolation required and containment isolation or ventilation valve(s) not closed when required AND Radiological release pathway to the environment exists 6. Inability to isolate any primary system discharging outside containment AND Radiological release pathway to the environment exists 7. Emergency Director judgment	1. Conditions requiring entry into Containment-RED path (CSP-Z.1) 2. Containment pressure ≥60 psig and rising (Containment-RED path, CSP-Z.1) 3. Hydrogen concentration in containment ≥6% 4. Containment pressure ≥25 psig with less than one train of containment spray and two containment accident fan cooler units operating 5. CET readings ≥1200°F (Core Cooling-RED path, CSP-C.1) AND Restoration procedures not effective with 15 min. 6. CET >700°F with RVLIS NR <25 ft and no RCPs running (Core Cooling-RED path, (CSP-C.1) AND Restoration procedures not effective within 15 min. 7. Containment rad monitor reading ≥15,900 R/hr 8. Emergency Director judgment

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SE	Alert	UE
1(2)RE 212	- - -	- - -	5.46E-2 µCi/cc	5.46E-4 µCi/cc
1(2)RE 305	- - -	- - -	5.46E-2 µCi/cc	5.46E-4 µCi/cc
1(2)RE 307	1.44E1 µCi/cc	1.44E0 µCi/cc	- - -	- - -
1(2)RE 309	1.44E1 µCi/cc	1.44E0 µCi/cc	- - -	- - -
RE 214	- - -	- - -	2.04E-2 µCi/cc	2.04E-4 µCi/cc
RE 315	- - -	- - -	2.04E-2 µCi/cc	2.04E-4 µCi/cc
RE 317	2.63E0 µCi/cc	2.63E-1 µCi/cc	- - -	- - -
RE 319	2.63E0 µCi/cc	2.63E-1 µCi/cc	- - -	- - -
1(2)RE 215	- - -	- - -	5.42E+2 µCi/cc	5.42E0 µCi/cc
RE 225	- - -	- - -	2.72E+2 µCi/cc	2.72E0 µCi/cc
RE 226	8.65E+3 µCi/cc	8.65E+2 µCi/cc	- - -	- - -
RE 221	- - -	- - -	3.16E-2 µCi/cc	3.16E-4 µCi/cc
RE 325	- - -	- - -	3.16E-2 µCi/cc	3.16E-4 µCi/cc
RE 327	4.34E0 µCi/cc	4.34E-1 µCi/cc	- - -	- - -
RE 224	- - -	- - -	4.18E-1 µCi/cc	4.18E-3 µCi/cc
1(2)RE 231/232	- - -	- - -	- - -	- - -
1(2)SG A/B				
1 ARV	6.69E-1 µCi/cc	6.69E-2 µCi/cc	N/A	N/A
1 SRV	2.48E-1 µCi/cc	2.48E-2 µCi/cc	N/A	N/A
2 SRV	1.24E-1 µCi/cc	1.24E-2 µCi/cc	N/A	N/A
3 SRV	8.25E-2 µCi/cc	8.25E-3 µCi/cc	N/A	N/A
4 SRV	6.20E-2 µCi/cc	6.20E-3 µCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.56E-3 µCi/cc	5.56E-5 µCi/cc
RE 230*	N/A	N/A	7.40E-2 µCi/cc	7.40E-4 µCi/cc

* with Waste Water Effluent discharge not isolated

Table H-1 Safe Shutdown Areas

- 1(2) Containment Building
- Primary Auxiliary Building
- Turbine Building
- Control Building
- Diesel Generator Building
- Gas Turbine Building
- Circ Water Pump House

Table R-2 Dose Proj. / Env. Measurement Class. Thresholds

	GE	SE	Alert
TEDE	1000 mRem	100 mRem	10 mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr. of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

HOT CONDITIONS (RCS > 200°F)

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | D Defueled

NMC — PBNP Emergency Action Level Matrix
EPIP 1.2, Emergency Classification

Approved: ____________ Manager Emergency Preparedness ____________ Date

COLD CONDITIONS

		GENERAL EMERGENCY	SITE EMERGENCY	ALERT	UNUSUAL EVENT
Loss of AC Power Sources		NONE	NONE	MA8.2 5 6 D Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥ 15 min.	MU8.1 5 6 D Unplanned loss of offsite AC power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥ 15 min.
Loss of DC Power Sources		NONE	NONE	NONE	MU9.1 5 6 ≤105 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for ≥ 15 min. due to unplanned activities
RCS	Temperature	NONE	NONE	MA13.1 5 6 An unplanned event results in RCS temperature exceeding 200°F for ≥ Table M-1 duration	MU13.1 5 6 An unplanned event results in RCS temperature ≥200°F **OR** Loss of all RCS temperature and Reactor Vessel level indication for ≥ 15 min.
RCS	Pressure	NONE	NONE	MA14.1 5 6 Unplanned RCS pressure rise ≥ 10 psig due to loss of decay heat removal	NONE
RCS	Level	MG15.1 5 6 1. Core uncovery for ≥ 30 min. as indicated by <u>EITHER</u> of the following: RVLIS NR ≤ [30 ft] 27 ft **OR** One or more of the following when Reactor Vessel water level cannot be monitored: - Containment High Range Radiation Monitor reading ≥ 10 R/hr - Erratic Source Range Monitor indication - Unexplained sump level rise **AND** 2. Containment challenged as indicated by one or more of the following: - Containment closre not established - Hydrogen concentration in containment 6% - Containment pressure ≥ 60 psig	MS15.1 5 6 *With containment closure <u>not</u> established,* RVLIS NR ≤ [33 ft] 30 ft **OR** With containment closure established, RVLIS NR ≤ [30 ft] 27 ft MS15.2 5 Reactor Vessel level cannot be monitored for ≥ 30 min. **AND** A loss of Reactor Vessel inventory as indicated by **<u>EITHER</u>**: Unexplained Containment Sump A level rise **OR** Erratic Source Range Monitor indication MS15.3 6 Reactor Vessel level cannot be monitored **AND** Indication of core uncovery as evidenced by one or more of the following: - Containment High Range Radiation Monitor reading ≥ 10 R/hr - Erratic Source Range Monitor indication - Unexplained Containment Sump A level increase	MA15.1 5 6 Loss of RCS or Reactor Vessel inventory as indicated by <u>EITHER</u>: LI-447 and LI-447A ≤ 0% when aligned **OR** If RCS or Reactor Vessel level cannot be monitored for ≥ 15 min., loss of inventory as indicated by unexplained Containment Sump A level rise	MU15.1 6 Unplanned RCS level lowering below 77.1% (1 foot below RPV flange) for ≥ 15min. **OR** If Reactor Vessel level cannot be monitored, loss of RPV inventory as indicated by unexplained Containment Sump A level rise

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SE	Alert	UE
1(2)RE 212	- - -	- - -	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 305	- - -	- - -	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 307	1.44E1 μCi/cc	1.44E0 μCi/cc	- - -	- - -
1(2)RE 309	1.44E1 μCi/cc	1.44E0 μCi/cc	- - -	- - -
RE 214	- - -	- - -	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 315	- - -	- - -	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 317	2.63E0 μCi/cc	2.63E-1 μCi/cc	- - -	- - -
RE 319	2.63E0 μCi/cc	2.63E-1 μCi/cc	- - -	- - -
1(2)RE 215	- - -	- - -	5.42E+2 μCi/cc	5.42E0 μCi/cc
RE 225	- - -	- - -	2.72E+2 μCi/cc	2.72E0 μCi/cc
RE 226	8.65E+3 μCi/cc	8.65E+2 μCi/cc	- - -	- - -
RE 221	- - -	- - -	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 325	- - -	- - -	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 327	4.34E0 μCi/cc	4.34E-1 μCi/cc	- - -	- - -
RE 224	- - -	- - -	4.18E-1 μCi/cc	4.18E-3 μCi/cc
1(2)RE 231/232	- - -	- - -	- - -	- - -
1(2)SG A/B				
1 ARV	6.69E-1 μCi/cc	6.69E-2 μCi/cc	N/A	N/A
1 SRV	2.48E-1 μCi/cc	2.48E-2 μCi/cc	N/A	N/A
2 SRV	1.24E-1 μCi/cc	1.24E-2 μCi/cc	N/A	N/A
3 SRV	8.25E-2 μCi/cc	8.25E-3 μCi/cc	N/A	N/A
4 SRV	6.20E-2 μCi/cc	6.20E-3 μCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.56E-3 μCi/cc	5.56E-5 μCi/cc
RE 230*	N/A	N/A	7.40E-2 μCi/cc	7.40E-4 μCi/cc

* with Waste Water Effluent discharge not isolated

Table R-2 Dose Proj. / Env. Measurement Class. Thresholds

	GE	SE	Alert
TEDE	1000 mRem	100 mRem	10 mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr. of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

Table M-1 RCS Reheat Duration Thresholds

Containment and RCS Barrier Status	Duration (min.)
1. RCS intact	60*
2. Containment closure established <u>AND EITHER</u>: RCS not intact **OR** RCS reduced inventory	20*
3. Containment closure <u>not</u> established **AND** RCS <u>not</u> intact	0

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is <u>not</u> applicable.

Table H-1 Safe Shutdown Areas

- 1(2) Containment Building
- Primary Auxiliary Building
- Turbine Building
- Control Building
- Diesel Generator Building
- Gas Turbine Building
- Circ Water Pump House

COLD CONDITIONS (RCS ≤ 200°F)

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | D Defueled

NMC

PBNP Emergency Action Level Matrix
EPIP 1.2, Emergency Classification

Approved: ____________ Manager Emergency Preparedness ____________ Date

EPIP 1.2.1

EMERGENCY ACTION LEVEL TECHNICAL BASIS

DOCUMENT TYPE:	Technical
CLASSIFICATION:	NNSR
REVISION:	0 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

__________ __________
__________ __________
__________ __________

TABLE OF CONTENTS

Category: Reactor Fuel **Sub-category:**Inadvertent Criticality

Initiating Condition: Inadvertent Criticality

EAL:

MU1.1	**Unusual Event**
An unplanned sustained positive startup rate observed on nuclear instrumentation.	

Mode Applicability:

3- Hot Standby, 4- Hot Shutdown, 5-Cold Shutdown, 6-Refuel

Basis:

This EAL 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 as well as inadvertent criticalities occurring in Hot Standby or Hot Shutdown mode. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification.

This condition can be identified using startup rate monitors (NI-31D/32D - Source Range Startup Rate, and NI-35D/36D - Intermediate Range Startup Rate). 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. The intent of "sustained" is to identify a critical condition.

Reference(s):

1. NEI CU8/SU8 - Inadvertent Criticality
2. OP 1B, Reactor Startup, Step 5.1 and 5.18.15

Category: Reactor Fuel **Sub-category:** Coolant Activity

Initiating Condition: Fuel Cladding Degradation

EAL:

MU2.1	**Unusual Event**
Coolant activity $\geq$0.8 µCi/gm dose equivalent I-131	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown, 5-Cold Shutdown, 6-Refuel

Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant technical specifications. Although the Tech Spec is applicable for modes 1, 2 and 3 (when $\geq$500 °F), it is appropriate that this EAL be applicable in all modes, as it indicates a potential degradation in the level of safety of the plant.

Reference(s):

1. NEI CU5/SU4 –Fuel Clad Degradation
2. Tech Spec 3.4.16, RCS Specific Activity, Unit 1 - Amendment No. 201 / Unit 2 Amendment No. 206

Category: Reactor Fuel **Sub-category:** Failed Fuel Monitor

Initiating Condition: Fuel Cladding Degradation

EAL:

MU3.1	**Unusual Event**
Failed Fuel Monitor (RE-109) ≥24 mRem/hr not due to a planned evolution.	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Elevated letdown line activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. Coolant activity at this value is well above normal expected values for this monitor. Any planned evolution for which increased values are expected do not apply. Fuel clad failure is a potential degradation of the level of safety and therefore warrants a declaration of an Unusual Event.

Reference(s):

1. NEI SU4 –Fuel Clad Degradation
2. Calc 96-0073, 2/29/96, (NEPG-86-515)
3. EPIP 10.2, Core Damage Estimation, Step 4.1

Category: Reactor Fuel **Sub-category:** Refueling Accidents & Other Radiation Monitors

Initiating Condition: Uncontrolled level drop in SFP

EAL:

MU4.1 **Unusual Event**

Spent fuel pool (reactor cavity during refueling) water level cannot be restored and maintained above the spent fuel pool low water level alarm setpoint

AND

Unplanned SFP Area Radiation Monitor readings rise

- o RE-105 SFP Area Low Range Area Radiation Monitor
- o RE-135 SFP Area High Range Area Radiation Monitor

Mode Applicability:

All

Basis:

The low level alarm is actuated by LC-634 at 62'-8" based on maintaining at least 6' of water on a withdrawn fuel assembly. Normal level is 63'-8". The definition of "... cannot be restored and maintained above..." allows the operator to visually observe the low water level condition, if possible, and to attempt water level restoration instructions as long as water level remains above the top of irradiated fuel.

When the fuel transfer canal is directly connected to the spent fuel pool and reactor cavity, there could exist the possibility of uncovering irradiated fuel in the fuel transfer canal. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the Reactor Vessel and spent fuel pool.

While a radiation monitor could detect a rise in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not the fuel will be or is uncovered. Elevated radiation monitor indications need to be combined with another indicator (or personnel report) of water loss.

This event escalates to an Alert if irradiated fuel outside the reactor vessel is uncovered via EAL# MA4.2.

Reference(s):

1. NEI AU2 – Unexpected Increase in Plant Radiation
2. DBD-13 Spent Fuel Pool Cooling and Filtration

Category: Reactor Fuel **Sub-category:** Refueling Accidents & Other Radiation Monitors

Initiating Condition: Radiation monitoring indicating damaged or uncovered irradiated fuel

EAL:

MA4.1 **Alert**

Confirmed sustained (10 minute average) alarm on any of the following radiation monitors resulting from an uncontrolled fuel handling process indicating damaged or uncovered irradiated fuel:

- o RE-105 SFP Area Low Range Area Radiation High Alarm ($\geq$10 mR/hr)
- o RE-135 SFP Area High Range Area Radiation High Alarm ($\geq$100 mR/hr)
- o 1(2) RE-211 Containment Air Particulate Monitor High Alarm ($\geq$0.5 μCi)
- o 1(2) RE-212B Containment Background Monitor High Alarm ($\geq$100 mR/hr)

Mode Applicability:

All

Basis:

This EAL addresses specific events that have resulted, or may result, in unexpected rises 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 degradation in the level of safety of the plant. These events escalate from EAL # MU4.1 in that fuel activity has been released or is anticipated due to fuel heatup. This EAL applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

When considering escalation, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, RP technicians)

This EAL is defined by the specific areas where irradiated fuel is located such as the reactor cavity, reactor vessel, or spent fuel pool.

A confirmed "uncontrolled fuel handling process" is defined as any event or activity related to the movement of irradiated fuel which results in unexpected or uncontrolled conditions. This terminology has been specifically added to exclude anticipated rises in area radiation levels as a result of actions performed in accordance with approved procedures during refueling operations.

The bases for the SFP area radiation high alarms and containment air particulate/background monitor high alarms are indicative of a fuel handling accident and are, therefore, appropriate for this EAL. While radiation monitors may detect a rise 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.

Reference(s):

1. NEI AA2 – Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel
2. PBNP RMSASRB
3. AOP-8B Irradiated Fuel Handling Accident in Containment
4. AOP-8C Fuel Handling Accident in PAB

Category: Reactor Fuel **Sub-category:** Refueling Accidents & Other Radiation Monitors

Initiating Condition: Indication of irradiated fuel uncovery

EAL:

MA4.2	**Alert**
Report of visual observation of irradiated fuel uncovered **OR** Loss of refueling water inventory as indicted by excessive makeup rate or unexpected lowering in refueling water storage tank level	

Mode Applicability:

All

Basis:

This EAL addresses specific events that have resulted, or may result, in unexpected rises 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 degradation in the level of safety of the plant. These events escalate from EAL MU4.1 in that fuel activity has been released or is anticipated due to fuel heatup. This EAL applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

When considering escalation, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, RP technicians)

This EAL is defined by the specific areas where irradiated fuel is located such as the reactor cavity, reactor vessel, or spent fuel pool.

There is no remote level indication that water level in the spent fuel pool or refueling cavity has dropped to the level of the fuel other than by visual observation. Since there is no level indicating system in the fuel transfer canal, visual observation of loss of water level would also be required.

Reference(s):

1. NEI AA2 – Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel

Category: RCS Leakage **Sub-category:** N/A

Initiating Condition: RCS Leakage

EAL:

MU5.1	**Unusual Event**
Unidentified or pressure boundary leakage ≥10 gpm	
<u>OR</u>	
Identified leakage ≥25 gpm	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown, 5-Cold Shutdown

Basis:

The conditions of this EAL 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. Positive indications in the control room of Reactor Coolant System (RCS) leakage to the containment are provided by equipment which permits continuous monitoring of containment air activity and humidity, and of runoff from the air recirculation units and containment floor drains to containment Sump A. This equipment provides indication of normal background radiation, which is indicative of a basic level of leakage from primary systems and components. Any rise in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The 10 gpm value for the unidentified leakage and pressure boundary leakage was selected because it is quantifiable with normal Control Room leak detection methods. 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 value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage. RCS leakage at these thresholds is well above the Tech Spec limits and is indicative of unsuccessful mitigation by the LCO Action Requirements.

Reference(s):

1. NEI CU1/SU5 – RCS Leakage
2. OP4A, Filling and Venting Reactor Coolant System, Step 5.6
3. TS 3.4.13, RCS Operational Leakage limits
4. OI-55, Primary Leak Rate Calculation
5. OM 3.19, Reactor Coolant System Leakage Determination

Category: Radioactivity Release / Area Radiation **Sub-category:** Effluent Monitors

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

EAL:

RU1.1 **Unusual Event**

Loss of control of radioactive materials as indicated by a valid reading on any monitors listed in Table R-1 column "UE" for ≥60 min. unless sample analysis can confirm release rates ≤ 2 x ODCM limits within this time period

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SE	Alert	UE
1(2)RE 212	---	---	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 305	---	---	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 307	1.44E1 μCi/cc	1.44E0 μCi/cc	---	---
1(2)RE 309	1.44E1 μCi/cc	1.44E0 μCi/cc	---	---
RE 214	---	---	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 315	---	---	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 317	2.63E0 μCi/cc	2.63E-1 μCi/cc	---	---
RE 319	2.63E0 μCi/cc	2.63E-1 μCi/cc	---	---
1(2)RE 215	---	---	5.42E+2 μCi/cc	5.42E0 μCi/cc
RE 225	---	---	2.72E+2 μCi/cc	2.72E0 μCi/cc
RE 226	8.65E+3 μCi/cc	8.65E+2 μCi/cc	---	---
RE 221	---	---	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 325	---	---	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 327	4.34E0 μCi/cc	4.34E-1 μCi/cc	---	---
RE 224	---	---	4.18E-1 μCi/cc	4.18E-3 μCi/cc
1(2)RE 231/232 1(2)SG A/B	----	----	----	----
1 ARV	6.69E-1 μCi/cc	6.69E-2 μCi/cc	N/A	N/A
1 SRV	2.48E-1 μCi/cc	2.48E-2 μCi/cc	N/A	N/A
2 SRV	1.24E-1 μCi/cc	1.24E-2 μCi/cc	N/A	N/A
3 SRV	8.25E-2 μCi/cc	8.25E-3 μCi/cc	N/A	N/A
4 SRV	6.20E-2 μCi/cc	6.20E-3 μCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.56E-3 μCi/cc	5.56E-5 μCi/cc
RE 230*	N/A	N/A	7.40E-2 μCi/cc	7.40E-4 μCi/cc

* with Waste Water Effluent discharge not isolated

Mode Applicability:

All

Basis:

Valid means that a radiation monitor reading has been confirmed by operators to be correct. Unplanned releases in excess of two times the site technical specifications that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not terminated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times T/S for 30 minutes does not exceed this initiating condition. Further, 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 will likely exceed 60 minutes and cannot be terminated.

The values shown for each monitor under column "UE" are approximately two times the calculated alarm setpoints (ODCM release limits) as specified in the RMSASRB choosing the highest values for variable conditions. In accordance with the requirements of 10 CFR 20.1001 and 20.2402 and Technical Specifications stated in Section 5 of the Radiological Effluent Control Manual, the alarm or trip setpoint for effluent monitors shall be established to annunciate at radiation levels which would result in an unrestricted area concentration equal to or less than the applicable maximum effluent concentration (MEC) for a single radionuclide. The appropriate detailed response to an effluent alarm is described in the PBNP RMS Alarm Set Point and Response Book.

Reference(s):

1. NEI AU1 – 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
2. PBNP ODCM Tables 2-1 and 2-2, Figures 2-1 and 2-2 and Table 3.9-2
3. EPIP 1.3 Dose Assessment and Protective Action Recommendations
4. RMS Alarm Set Point and Response Book (RMSASRB)
5. STPT 13.4, Radiation Monitoring System: Effluent Monitors

Category: Radioactivity Release / Area Radiation **Sub-category:** Effluent Monitors

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

EAL:

RA1.1 **Alert**

Loss of control of radioactive materials as indicated by a valid reading on any monitors listed in Table R-1 column "Alert" for ≥15 min. unless sample analysis can confirm release rates ≤ 200 x ODCM limits within this time period

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SE	Alert	UE
1(2)RE 212	---	---	5.46E-2 µCi/cc	5.46E-4 µCi/cc
1(2)RE 305	---	---	5.46E-2 µCi/cc	5.46E-4 µCi/cc
1(2)RE 307	1.44E1 µCi/cc	1.44E0 µCi/cc	---	---
1(2)RE 309	1.44E1 µCi/cc	1.44E0 µCi/cc	---	---
RE 214	---	---	2.04E-2 µCi/cc	2.04E-4 µCi/cc
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RE 317	2.63E0 µCi/cc	2.63E-1 µCi/cc	---	---
RE 319	2.63E0 µCi/cc	2.63E-1 µCi/cc	---	---
1(2)RE 215	---	---	5.42E+2 µCi/cc	5.42E0 µCi/cc
RE 225	---	---	2.72E+2 µCi/cc	2.72E0 µCi/cc
RE 226	8.65E+3 µCi/cc	8.65E+2 µCi/cc	---	---
RE 221	---	---	3.16E-2 µCi/cc	3.16E-4 µCi/cc
RE 325	---	---	3.16E-2 µCi/cc	3.16E-4 µCi/cc
RE 327	4.34E0 µCi/cc	4.34E-1 µCi/cc	---	---
RE 224	---	---	4.18E-1 µCi/cc	4.18E-3 µCi/cc
1(2)RE 231/232 1(2)SG A/B		----	----	----
1 ARV	6.69E-1 µCi/cc	6.69E-2 µCi/cc	N/A	N/A
1 SRV	2.48E-1 µCi/cc	2.48E-2 µCi/cc	N/A	N/A
2 SRV	1.24E-1 µCi/cc	1.24E-2 µCi/cc	N/A	N/A
3 SRV	8.25E-2 µCi/cc	8.25E-3 µCi/cc	N/A	N/A
4 SRV	6.20E-2 µCi/cc	6.20E-3 µCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.56E-3 µCi/cc	5.56E-5 µCi/cc
RE 230*	N/A	N/A	7.40E-2 µCi/cc	7.40E-4 µCi/cc

* with Waste Water Effluent discharge not isolated

Mode Applicability:

All

Basis:

This EAL addresses a potential or actual lowering in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. PBNP incorporates features intended to control the release of radioactive effluents to the environment. Additionally, 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). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of degradation in these features and/or controls.

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 will likely exceed 15 minutes and cannot be terminated. 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.

This event escalates from the Unusual Event by escalating the magnitude of the release by a factor of 100.

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

Reference(s):

1. NEI AA1 – 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
2. PBNP ODCM Tables 2-1 and 2-2 and Figures 2-1 and 2-2
3. EPIP 1.3 Dose Assessment and Protective Action Recommendations
4. RMS Alarm Setpoint and Response Book (RMSASRB)
5. STPT 13.4

Category: Radioactivity Release / Area Radiation **Sub-category:** Effluent Monitors

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

EAL:

RS1.1 **Site Emergency**

A valid reading on any monitors listed in Table R-1 column "SE" for ≥15 min. unless dose assessment can confirm releases are below Table R-2 column "SE" within this time period

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SE	Alert	UE
1(2)RE 212	---	---	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 305	---	---	5.46E-2 μCi/cc	5.46E-4 μCi/cc
1(2)RE 307	1.44E1 μCi/cc	1.44E0 μCi/cc	---	---
1(2)RE 309	1.44E1 μCi/cc	1.44E0 μCi/cc	---	---
RE 214	---	---	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 315	---	---	2.04E-2 μCi/cc	2.04E-4 μCi/cc
RE 317	2.63E0 μCi/cc	2.63E-1 μCi/cc	---	---
RE 319	2.63E0 μCi/cc	2.63E-1 μCi/cc	---	---
1(2)RE 215	---	---	5.42E+2 μCi/cc	5.42E0 μCi/cc
RE 225	---	---	2.72E+2 μCi/cc	2.72E0 μCi/cc
RE 226	8.65E+3 μCi/cc	8.65E+2 μCi/cc	---	---
RE 221	---	---	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 325	---	---	3.16E-2 μCi/cc	3.16E-4 μCi/cc
RE 327	4.34E0 μCi/cc	4.34E-1 μCi/cc	---	---
RE 224	---	---	4.18E-1 μCi/cc	4.18E-3 μCi/cc
1(2)RE 231/232 1(2)SG A/B	----	----	----	----
1 ARV	6.69E-1 μCi/cc	6.69E-2 μCi/cc	N/A	N/A
1 SRV	2.48E-1 μCi/cc	2.48E-2 μCi/cc	N/A	N/A
2 SRV	1.24E-1 μCi/cc	1.24E-2 μCi/cc	N/A	N/A
3 SRV	8.25E-2 μCi/cc	8.25E-3 μCi/cc	N/A	N/A
4 SRV	6.20E-2 μCi/cc	6.20E-3 μCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.56E-3 μCi/cc	5.56E-5 μCi/cc
RE 230*	N/A	N/A	7.40E-2 μCi/cc	7.40E-4 μCi/cc

* with Waste Water Effluent discharge not isolated

Table R-2 Dose Projection / Env. Measurement Classification Thresholds

	GE	SE	ALERT
TEDE	1000 mRem	100 mRem	10mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

Mode Applicability:

All

Basis:

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed a fraction (10%) 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 EALs, this EAL 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 Table R-2 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-1 column "SE" effluent monitor readings have been determined using WEDAP by back calculating from the dose values specified in Table R-2. The back calculations were performed using default assumptions and based on annual average meteorology. With the exception of RE 231 and RE 232 (Steamline Vent) the source term was based on LOCA/GAP release in containment with filtration where applicable.

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 readings listed in Table R-1.

Reference(s):

1. NEI AS1 – 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
2. WEDAP Sensitivity Runs
3. EPIP 1.3 Dose Assessment and Protective Action Recommendations
4. FSAR Table 2.6-3 Stability Index Distribution
5. FSAR Table 2.6-4 Site Atmospheric Stability Analysis Annual Average 13 month Data
6. FSAR Figure 2.6-2 Stability Class Distribution in Percent of Total Observed
7. DBD-T-46 Section 3.1 Station Blackout

Category: Radioactivity Release / Area Radiation **Sub-category:** Effluent Monitors

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

EAL:

RG1.1 **General Emergency**

A valid reading on any monitors listed in Table R-1 column "GE" for ≥15 min. unless dose assessment can confirm releases are below Table R-2 column "GE" within this time period

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SE	Alert	UE
1(2)RE 212	---	---	5.46E-2 µCi/cc	5.46E-4 µCi/cc
1(2)RE 305	---	---	5.46E-2 µCi/cc	5.46E-4 µCi/cc
1(2)RE 307	1.44E1 µCi/cc	1.44E0 µCi/cc	---	---
1(2)RE 309	1.44E1 µCi/cc	1.44E0 µCi/cc	---	---
RE 214	---	---	2.04E-2 µCi/cc	2.04E-4 µCi/cc
RE 315	---	---	2.04E-2 µCi/cc	2.04E-4 µCi/cc
RE 317	2.63E0 µCi/cc	2.63E-1 µCi/cc	---	---
RE 319	2.63E0 µCi/cc	2.63E-1 µCi/cc	---	---
1(2)RE 215	---	---	5.42E+2 µCi/cc	5.42E0 µCi/cc
RE 225	---	---	2.72E+2 µCi/cc	2.72E0 µCi/cc
RE 226	8.65E+3 µCi/cc	8.65E+2 µCi/cc	---	---
RE 221	---	---	3.16E-2 µCi/cc	3.16E-4 µCi/cc
RE 325	---	---	3.16E-2 µCi/cc	3.16E-4 µCi/cc
RE 327	4.34E0 µCi/cc	4.34E-1 µCi/cc	---	---
RE 224	---	---	4.18E-1 µCi/cc	4.18E-3 µCi/cc
1(2)RE 231/232 1(2)SG A/B	---	---	----	----
1 ARV	6.69E-1 µCi/cc	6.69E-2 µCi/cc	N/A	N/A
1 SRV	2.48E-1 µCi/cc	2.48E-2 µCi/cc	N/A	N/A
2 SRV	1.24E-1 µCi/cc	1.24E-2 µCi/cc	N/A	N/A
3 SRV	8.25E-2 µCi/cc	8.25E-3 µCi/cc	N/A	N/A
4 SRV	6.20E-2 µCi/cc	6.20E-3 µCi/cc	N/A	N/A
1(2)RE 229	N/A	N/A	5.56E-3 µCi/cc	5.56E-5 µCi/cc
RE 230*	N/A	N/A	7.40E-2 µCi/cc	7.40E-4 µCi/cc

* with Waste Water Effluent discharge not isolated

Table R-2 Dose Projection / Env. Measurement Classification Thresholds

	GE	SE	ALERT
TEDE	1000 mRem	100 mRem	10mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

Mode Applicability:

All

Basis:

This EAL 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 EALs, this EAL 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 Table R-2 TEDE dose is set at the EPA PAG, while the 5000 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-1 column "GE" effluent monitor readings have been determined using WEDAP by back calculating from the dose values specified in Table R-2. The back calculations were performed using default assumptions and based on annual average meteorology. With the exception of RE 231 and RE 232 (Steamline Vent) the source term was based on LOCA/GAP release in containment with filtration where applicable.

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 readings listed in Table R-1.

Reference(s):

1. NEI 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
2. WEDAP Sensitivity Runs
3. EPIP 1.3 Dose Assessment and Protective Action Recommendations
4. FSAR Table 2.6-3 Stability Index Distribution
5. FSAR Table 2.6-4 Site Atmospheric Stability Analysis Annual Average 13 month Data
6. FSAR Figure 2.6-2 Stability Class Distribution in Percent of Total Observed
7. DBD-T-46 Section 3.1 Station Blackout

Category: Radioactivity Release / Area Radiation

Sub-category: Dose Projections / Environmental Measurements / Release Rates

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

EAL:

RU2.1	**Unusual Event**
An unplanned gaseous or liquid release with rates $\geq$2 x ODCM limits for $\geq$60 min.	

Mode Applicability:

All

Basis:

Unplanned releases (without a discharge permit or exceeding the limits of a discharge permit) in excess of two times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times T/S for 30 minutes does not exceed this initiating condition. Further, 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.

Reference(s):

1. NEI AU1 – 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
2. PBNP ODCM

Category: Radioactivity Release / Area Radiation

Sub-category: Dose Projections / Environmental Measurements / Release Rates

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

EAL:

RA2.1	**Alert**
An unplanned gaseous or liquid release with rates ≥200 x ODCM limits for ≥15 min.	

Mode Applicability:

All

Basis:

Confirmed sample analyses of unplanned releases (without or exceeding the limits of a discharge permit) in excess of two hundred times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, an actual or potential substantial degradation in the level of safety. This event escalates from the Unusual Event by raising the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10CFR20 non-occupational MPC limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mRem/hr. The required release duration was reduced to 15 minutes in recognition of the raised severity.

Reference(s):

1. NEI AA1 – 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
2. PBNP ODCM Tables 2-1 and 2-2, Figures 2-1 and 2-2 and Table 3.9-2
3. EPIP 1.3 Dose Assessment and Protective Action Recommendations

Category: Radioactivity Release / Area Radiation

Sub-category: Dose Projections / Environmental Measurements / Release Rates

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

EAL:

RA2.2 **Alert**

Dose projections or field surveys resulting from an unplanned actual or imminent release which indicate doses / dose rates ≥ Table R-2 column "Alert" at the site boundary or beyond.

Table R-2 Dose Projection / Env. Measurement Classification Thresholds

	GE	SE	ALERT
TEDE	1000 mRem	100 mRem	10mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

Mode Applicability:

All

Basis:

Offsite integrated doses in excess of 10 mRem TEDE or dose rates in excess of 10 mRem/hr TEDE represent an uncontrolled situation and hence, an actual or potential substantial degradation in the level of safety. This event escalates from the Unusual Event by raising the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times Technical Specifications). Prorating the 500 mRem/yr basis of 10CFR20 for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be 10 mRem/hr.
The 'site boundary' is defined by an approximately 1 mile radius around the site Protected Area.

Reference(s):

1. NEI AA1 – 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
2. PBNP ODCM Tables 2-1 and 2-2, Figures 2-1 and 2-2 and Table 3.9-2
3. EPIP 1.3 Dose Assessment and Protective Action Recommendations

Category: Radioactivity Release / Area Radiation

Sub-category: Dose Projections / Environmental Measurements / Release Rates

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

EAL:

RS2.1 **Site Emergency**

Dose projections or field surveys resulting from actual or imminent release which indicate doses / dose rates ≥ Table R-2 column "SE" at the site boundary or beyond

Table R-2 Dose Projection / Env. Measurement Classification Thresholds

	GE	SE	ALERT
TEDE	1000 mRem	100 mRem	10mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

Mode Applicability:

All

Basis:

The 100 mRem integrated TEDE dose in this EAL is based on the 10CFR20 annual average population exposure and is indicative of an actual or likely major failure of plant functions needed for the protection of the public. This value also provides a desirable gradient (one order of magnitude) between the Alert, Site Emergency and General Emergency classes. Exposures less than this limit are not consistent with the Site Emergency class description. The 500 mRem integrated CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guides for TEDE and thyroid exposure.
In establishing the dose rate emergency action levels, a duration of one hour is assumed. Therefore, the dose rate EALs are based on a site boundary dose rate of 100 mRem/hr TEDE or 500 mRem/hr CDE thyroid, whichever is more limiting. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

The terminology used in Table R-2 "External exposure rate" is intended to equate to the CEDE rate specified in EPIP 1.3 Dose Assessment and Protective Action Recommendations. The term "Thyroid exposure rate (for one hour of inhalation)" equates to the CDE thyroid exposure rate specified in EPIP 1.3 Dose Assessment and Protective Action Recommendations.
The 'site boundary' is defined by an approximately 1 mile radius around the site Protected Area.

Reference(s):

1. NEI AS1 – 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
2. EPIP 1.3 Dose Assessment and Protective Action Recommendations

Category: Radioactivity Release / Area Radiation

Sub-category: Dose Projections / Environmental Measurements / Release Rates

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

EAL:

RG2.1 **General Emergency**

Dose projections or field surveys resulting from actual or imminent release which indicate doses / dose rates ≥ Table R-2 column "GE" at the site boundary or beyond.

Table R-2 Dose Projection / Env. Measurement Classification Thresholds

	GE	SE	ALERT
TEDE	1000 mRem	100 mRem	10mRem
CDE Thyroid	5000 mRem	500 mRem	N/A
External exposure rate	1000 mRem/hr	100 mRem/hr	10 mRem/hr
Thyroid exposure rate (for 1 hr of inhalation)	5000 mRem/hr	500 mRem/hr	N/A

Mode Applicability:
All

Basis:

The General Emergency values of Table R-2 are based on the boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mRem TEDE or 5000 mRem CDE thyroid for the actual or projected duration of the release. The 1000 mRem TEDE and the 5000 mRem CDE thyroid integrated dose are based on the EPA protective action guidance which indicates that public protective actions are indicated if the dose exceeds 1 rem TEDE or 5 rem CDE thyroid. This is consistent with the emergency class description for a General Emergency in that it is indicative of substantial core degradation or melting and loss of containment integrity. This level constitutes the upper level of the desirable gradient for the Site Emergency.

Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible. In establishing the dose rate emergency action levels, a duration of one hour is assumed. Therefore, the dose rate EALs are based on a site boundary dose rate of 1000 mRem/hr TEDE or 5000 mRem/hr CDE thyroid, whichever is more limiting.

The terminology used in Table R-2 "External exposure rate" is intended to equate to the CEDE rate specified in EPIP 1.3 Dose Assessment and Protective Action Recommendations. The term "Thyroid exposure rate (for one hour of inhalation)" equates to the CDE thyroid exposure rate specified in EPIP 1.3 Dose Assessment and Protective Action Recommendations.

The 'site boundary' is defined by an approximately 1 mile radius around the site Protected Area.

Reference(s):

1. NEI 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
2. EPIP 1.3 Dose Assessment and Protective Action Recommendations
3. FSAR Volume 1 Figure 2.2-3 Site Topography Map

Category: Radioactivity Release / Area Radiation

Sub-category: Area Radiation Levels

Initiating Condition: Unexpected rise in plant radiation

EAL:

RU3.1	**Unusual Event**
Any sustained (10 minute average) direct Area Rad Monitor readings ≥100 x alarm or offscale high not resulting from a planned event or evolution	

Mode Applicability:

All

Basis:

Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.
This EAL addresses rises in radiation levels inside the plant not due to planned events or evolutions. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant. Area radiation levels above 100 times the alarm setpoint have been selected because they are readily identifiable on Area Rad Monitor instrumentation. Since Area Rad Monitor setpoints are nominally set approximately one decade over normal levels, 100 times the alarm setpoint provides an appropriate threshold for emergency classification. 100 times the alarm setpoint is, therefore, approximately 1000 times the normal level. For those Area Rad Monitors whose upper range limit is less than 100 times the alarm setpoint, a value of offscale high is used. This EAL escalates to an Alert, if the elevated radiation levels impair the level of safe plant operation.

Reference(s):

1. AU2 – Unexpected increase in plant radiation
2. PBNP RMSASRB

Category: Radioactivity Release / Area Radiation

Sub-category: Area Radiation Levels

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

EAL:

RA3.1 **Alert**

Sustained (10 minute average) area radiation levels $\geq$15 mR/hr in **EITHER**:

Control Room (RE 101)

OR

Central Alarm Station (by survey)

OR

Secondary Alarm Station (by survey)

Mode Applicability:

All

Basis:

This EAL addresses elevated radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. 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. The value of 15 mR/hr is derived from the General Design Criteria (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. A 30 day duration implies an event potentially more significant than an Alert.

This EAL is not intended to apply to anticipated temporary radiation rises due to planned events or evolutions (e.g., radwaste container movement, depleted resin transfers, etc.).

It is the 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 or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the elevated radiation levels and determine if any other EALs may be involved.

Reference(s):

1. NEI AA3 – Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown
2. GDC 19
3. NUREG-0737, "Clarification of TMI Action Plan Requirements", Section III.D.3
4. PBNP RMSASRB

Category: Radioactivity Release / Area Radiation

Sub-category: Area Radiation Levels

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

EAL:

RA3.2	**Alert**
Sustained (10 minute average) abnormal area radiation levels ≥12 R/hr in any Table H-1 Safe Shutdown Area **AND** Access to affected area is required for safe operation or shutdown	

Mode Applicability:

All

Basis:

This EAL addresses elevated radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Area radiation levels at or above 12 R/hr are indicative of radiation fields which may limit personnel access or adversely affect equipment whose operation may be needed to assure adequate core cooling or shutdown the reactor. The basis of the 12 R/hr value is as follows:

> The PBNP annual administrative personnel exposure limit is 2 Rem/Year. Assuming an emergency worker is at his administrative limit, any emergency worker needing access to a plant area for the safe shutdown of the plant could receive up to an additional 3 Rem without exceeding the legal 10CFR20 annual exposure limit of 5 Rem and thus the need for emergency exposure authorization. Assuming that an activity required to be performed in the plant would, on average, require a 15 minute stay time in that area, an area exposure rate of 12 R/hr would not unduly restrict access to areas necessary for safe plant shutdown.

It is the 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 or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the elevated radiation levels and determine if any other EAL may be involved.

This EAL is not meant to apply to rises in the containment radiation monitors as these are events that are addressed in other EALs. Nor is it intended to apply to anticipated temporary radiation rises due to planned evolutions or events (e.g., radwaste container movement, depleted resin transfers, etc.).

Reference(s):

1. NEI AA3 – Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown
2. EPIP-5.1 Personnel Emergency Dose Authorization

Category: Control Room Evacuation **Sub-category:** N/A

Initiating Condition: Control Room Evacuation Has Been Initiated

EAL:

HA5.1	**Alert**
Entry into AOP-10 Control Room Inaccessibility due to Control Room Evacuation	

Mode Applicability:

All

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are 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 Emergency via HS5.1.

Reference(s):

1. NEI HA5 – Control Room Evacuation Has Been Initiated
2. AOP-10 Control Room Inaccessibility
3. AOP-10A Safe Shutdown - Local Control
4. AOP-10B Safe to Cold Shutdown in Local Control

Category: Control Room Evacuation **Sub-category:**N/A

Initiating Condition: Control Room evacuation has been initiated and plant control cannot be established

EAL:

HS5.1	**Site Emergency**
Control Room evacuation **<u>AND</u>** Transfer of reactivity, RCS inventory and secondary heat removal control functions cannot be established per AOP-10A Safe Shutdown - Local Control in ≤ 15 min.	

Mode Applicability:

All

Basis:

This EAL indicates that expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. The intent of this EAL is to capture events in which control of the plant cannot be reestablished in a timely manner.

Once the Control Room is evacuated the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis is placed on 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). In Cold Shutdown and Refuel modes, operator concern is directed toward maintaining core cooling such as is discussed in Generic Letter 88-17, "Loss of Decay Heat Removal." In Power Operation, and Hot Shutdown modes, operator concern is primarily directed toward maintaining critical safety functions and thereby assuring fission product barrier integrity.

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

Reference(s):

1. NEI HS2 – Control Room evacuation has been initiated and plant control cannot be established
2. AOP-10 Control Room Inaccessibility
3. AOP-10A Safe Shutdown - Local Control
4. AOP-10B Safe to Cold Shutdown in Local Control

Category: Communication Loss **Sub-category:** N/A

Initiating Condition: Unplanned loss of all onsite or offsite communications capabilities

EAL:

MU6.1 **Unusual Event**

Loss of all communications capability affecting the ability to **EITHER**:

Perform routine operations

OR

Notify offsite agencies or personnel

Mode Applicability:

All

Basis:

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems with offsite authorities. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10CFR50.72.

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

Onsite/offsite communications include one or more of the systems listed in Table M-2.

Table M-2 Communications Systems		
System	**Onsite**	**Offsite**
Gai-tronics	X	
Security Radio	X	
Internal Commercial PBX system	X	
Portable radios via two radio base stations and four radio desk sets	X	
Two-Digit Dial-Select phone system	X	
Sound power phones	X	
Emergency Notification System (ENS)		X
Health Physics Network (HPN)		X
Operations Control Counterpart Link (OCCL)		X
Management Counterpart Link (MCL)		X
Protective Measures Counterpart Link (PMCL)		X
Reactor Safety Counterpart Link (RSCL)		X
Nuclear Accident Reporting System (NARS)		X
PBX System		X
General Telephone Lines		X
Manitowoc City Sheriff's Department FM Radio		X

Reference(s):

1. CU6 / SU6 – Unplanned loss of all onsite or offsite communications capabilities
2. EPMP 2.1, Testing of Communications Equipment
3. EPMP 2.1A, Monthly Communications Test

Category: Hazards **Sub-category:** Security Threats

Initiating Condition: Confirmed security event which indicates a potential degradation in the level of safety of the plant

EAL:

HU1.1 **Unusual Event**

Indication of attempted sabotage, hostage/extortion, civil disturbance or strike action onsite

OR

Notification of any credible site-specific threat by the Security Shift Supervisor or outside agency (NRC, military or law enforcement)

Mode Applicability:

All

Basis:

This EAL is based on the PBNP Security and Safeguards Contingency 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.

The second condition is included to ensure the appropriate notifications for the security threat are made in a timely manner. The determination of "credible" is made through the use of information found in the Security and Safeguards Contingency Plan. Only the plant or site to which the specific threat is made need declare the Unusual Event. Threats made that are ambiguous or are not unit-specific (e.g. "the PBNP site") may be conservatively interpreted to include both the units. This would result in an emergency classification at both Unit 1 and Unit 2. Reference is made to the Security Shift Supervisor because this individual is the designated on-site person who is 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 and Safeguards Contingency Plan.

Intrusion into the site Protected Area by an adversary would result in escalation to an Alert under EAL# HA1.1.

Reference(s):

1. NEI HU4 – Confirmed security event which indicates a potential degradation in the level of safety of the plant
2. NRC Safeguards Advisory 10/6/01
3. PBNP Security And Safeguards Contingency Plan
4. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
5. NMC fleet Security Threat Assessment Policy, SE 0018

Category: Hazards **Sub-category:** Security Threats

Initiating Condition: Confirmed security event in a site Protected Area

EAL:

HA1.1	**Alert**
Intrusion into the site Protected Area by an adversary indicated by notification by the Security Shift Supervisor to implement AOP-29 for a PA intrusion.	

Mode Applicability:

All

Basis:

An adversary is an armed or suspected to be armed intruder whose intent is to commit sabotage, disrupt station operations or otherwise commit a crime on station property. A confirmed intrusion report is satisfied if physical evidence indicates the presence of an adversary within the Protected Area (i.e., confirmed explosive device).

The Security and Safeguards Contingency 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 an adversary would result in escalation to a Site Emergency under EAL# HS1.1.

Reference(s):

1. NEI HA4 – Confirmed security event in a plant Protected Area
2. Security And Safeguards Contingency Plan
3. AOP-29, Security Threat
4. NMC fleet Security Threat Assessment Policy, SE 0018

Category: Hazards **Sub-category:** Security Threats

Initiating Condition: Confirmed security event in a plant Vital Area

EAL:

HS1.1	**Site Emergency**
Intrusion into a Vital Area by an adversary indicated by notification by the Security Shift Supervisor to implement AOP-29 for Vital Area intrusion.	

Mode Applicability:

All

Basis:

An adversary is an armed or suspected to be armed intruder whose intent is to commit sabotage, disrupt station operations or otherwise commit a crime on station property. A confirmed intrusion report is satisfied if physical evidence indicates the presence of an adversary within the security Vital Area (i.e., confirmed explosive device).

Consideration is given to the following events when evaluating an event against the criteria of the site Security And Safeguards Contingency Plan: sabotage, bomb threat and hostage/extortion. The Security And Safeguards Contingency Plan identifies numerous events/conditions that constitute a threat/compromise to a station security. Only 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 result in escalation to a General Emergency per EAL # HG1.1.

Reference(s):

1. NEI HS1 – Confirmed security event in a plant Vital Area
2. Security And Safeguards Contingency Plan
3. NMC fleet Security Threat Assessment Policy, SE 0018

Category: Hazards **Sub-category:** Security Threats

Initiating Condition: Security event resulting in loss of physical control of the facility

EAL:

HG1.1	**General Emergency**

An adversary has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions:

Reactivity control
RCS inventory
Secondary heat removal
Spent Fuel Pool integrity

Mode Applicability:

All

Basis:

This EAL encompasses conditions under which an adversary has taken physical control of plant vital areas (containing vital equipment or controls) 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, the EAL threshold is not met. Loss of physical control of the Control Room or remote shutdown capability alone may not prevent the ability to maintain safety functions.

This EAL also addresses loss of physical control of spent fuel pool cooling systems if imminent damage of fuel in the spent fuel pool is likely.

Reference(s):

1. NEI HG1 – Security event resulting in loss of physical control of the facility
2. Security And Safeguards Contingency Plan
3. NMC fleet Security Threat Assessment Policy, SE 0018

Category: Hazards **Sub-category:** Fire or Explosion

Initiating Condition: Fire within Protected Area Boundary not Extinguished within 15 Minutes of Detection

EAL:

HU2.1	**Unusual Event**
Confirmed fire in the Protected Area not extinguished in ≤ 15 minutes of Control Room notification.	

Mode Applicability:

All

Basis:

The purpose of this EAL is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, a confirmed fire is a fire that has been identified through visual observation and report by plant personnel, or sensor alarm indication. The 15-minute period begins when a credible report is received that a fire is occurring or a valid fire detection system alarm is received. Validation of a fire detection system alarm includes actions that can be taken within the control room or other nearby location to ensure that the alarm is not spurious. A validated alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to validate the alarm.

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished. This excludes fires within administration buildings, waste paper basket fires and other small fires of no safety consequence.

EAL# HA2.1 provides escalation to the Alert classification.

Reference(s):

1. NEI HU2 – Fire within Protected Area boundary not extinguished within 15 minutes of detection
2. PBNP FSAR Table 3.3-1
3. Bechtel Drawing C-3 Plant Areas

Category: Hazards **Sub-category:** Fire or Explosion

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU2.2	**Unusual Event**
Report by plant personnel of an explosion within Protected Area resulting in visible damage to permanent structures or equipment	

Mode Applicability:

All

Basis:

For this EAL, only those explosions of sufficient force to visibly damage permanent structures or equipment within the Protected Area should be considered. An explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized equipment that imparts significant energy to nearby structures or equipment. 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 (e.g., deformation, scorching, etc.) is sufficient for declaration. The Emergency Director also needs to consider any security aspects of the explosion.

Reference(s):

1. HU1 – Natural or destructive phenomena affecting the Protected Area
2. Bechtel Drawing C-3 Plant Areas

Category: Hazards **Sub-category:** Fire or Explosion

Initiating Condition: Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown

EAL:

HA2.1 **Alert**

Fire or explosion in any Table H-1 Safe Shutdown Area, which results in **EITHER**:

Visible damage to plant equipment or structures needed for safe shutdown

OR

Affected safety system performance is degraded indicating damage to a safety system

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

Mode Applicability:

All

Basis:

Table H-1 lists areas that 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. An explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized equipment that imparts significant energy to nearby structures or equipment.

Explosions of sufficient force are those that cause visible damage to permanent structures or equipment required for safe operation, or result in degraded performance of safety systems without visible damage within the identified plant areas. No attempt is made to assess the actual magnitude of the damage.

The wording of this EAL does not imply that a quantitative assessment of safety system performance should be performed; rather that observation that degraded safety system parameters is a result of the event. Either a physical or functional determination of degraded performance is sufficient to classify this event.

The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform damage assessments. The Emergency Director also needs to consider the security aspects of the explosions.

Reference(s):

1. NEI HA2 – Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown
2. PSA Section 8.0 Fire Hazards Analysis Table 8.1.3-1 Safe Shutdown Systems

Category: Hazards **Sub-category:** Vehicle Crash/Toxic and Flammable Gas

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU3.1	**Unusual Event**
Vehicle crash into plant structures or systems within the Protected Area	

Mode Applicability:

All

Basis:

This EAL 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 a crash is confirmed to affect a plant Vital Area, the event may escalate to the Alert classification under EAL # HA3.1.

Reference(s):

1. NEI HU1 – Natural or destructive phenomena affecting the Protected Area
2. Bechtel Drawing C-3 Plant Area

Category: Hazards **Sub-category:** Vehicle Crash/Toxic and Flammable Gas

Initiating Condition: Release of toxic or flammable gases deemed detrimental to normal operation of the plant

EAL:

HU3.2	**Unusual Event**
Report or detection of toxic or flammable gases that could enter or have entered within the Protected Area in amounts that could affect the health of plant personnel or safe plant operation **OR** Report by local, county or state officials for evacuation or sheltering of site personnel based on an offsite event	

Mode Applicability:

All

Basis:

This EAL is based on the existence of uncontrolled releases of toxic or flammable gas affecting normal plant operations or the health of plant personnel. The release may have originated within the Protected Area, or it may have originated offsite and subsequently drifted inside the Protected Area. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on 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 EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect plant operations or personnel safety.

Should the release affect plant Vital Areas, escalation to an Alert would be based on EAL# HA3.2.

Should an explosion or fire occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL# HA2.1.

Reference(s):

1. NEI HU3 – Release of toxic or flammable gases deemed detrimental to normal operation of the plant
2. PBNP FSAR Table 3.3-1

Category: Hazards **Sub-category:** Vehicle Crash/Toxic and Flammable Gas

Initiating Condition: Natural or destructive phenomena affecting the plant Vital area

EAL:

HA3.1	**Alert**
Vehicle crash which precludes personnel access to or damages equipment in one or more Table H-1 Safe Shutdown Areas	

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

Mode Applicability:

All

Basis:

Personnel access to safe shutdown areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses vehicle crashes that preclude personnel access to safe shutdown areas or may have resulted in the area being subjected to forces beyond design limits. 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.

This EAL 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.

Reference(s):

1. NEI HA1 – Natural and destructive phenomena affecting the plant Vital Area
2. PSA Section 8.0 Fire Hazards Analysis Table 8.1.3-1 Safe Shutdown Systems

Category: Hazards **Sub-category:** Vehicle Crash/Toxic and Flammable Gas

Initiating Condition: Release of toxic or flammable gases within or contiguous to a Vital Area which jeopardizes operation of systems required to establish or maintain safe shutdown

EAL:

HA3.2 **Alert**

Report or detection of toxic or flammable gases within any Table H-1 Safe Shutdown Area in concentrations that **EITHER**:

Will be immediately life threatening to plant personnel

OR

Exceed the lower flammability limit

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

Mode Applicability:

All

Basis:

This EAL is based on toxic or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. This EAL applies to buildings and areas contiguous to safe shutdown areas or other significant buildings or areas. It is appropriate that more frequent monitoring be done to ascertain whether consequential damage has occurred.

The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Safe Shutdown Area or any area or building contiguous to Safe Shutdown Area. Exposure to an IDLH atmosphere will result in immediate harm to unprotected personnel, and would preclude access to any such affected areas.

The second condition is met when the flammable gas concentration in a Safe Shutdown Area or any building or area contiguous to a Safe Shutdown 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 condition 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.

Reference(s):

1. NEI HA3 – Release of toxic or flammable gases within or contiguous to a Vital Area which jeopardizes operation of systems required to establish or maintain safe shutdown
2. PSA Section 8.0 Fire Hazards Analysis Table 8.1.3-1 Safe Shutdown Systems

Category: Hazards **Sub-category:** Natural Events

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU4.1	**Unusual Event**

Activation of 2 or more seismic monitors (SEI 6210 through 6213)
AND
Verified by:
- Actual ground shaking

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

Mode Applicability:

All

Basis:

PBNP seismic monitors actuates at 0.01 g. Seismic monitors are located in the following areas:

SEI 6210 #3 Warehouse
SEI 6211 Unit 1 Facade
SEI 6212 Drum Prep. Room
SEI 6213 El 8' between vital switchgear room and AFW Tunnel

An earthquake "felt" and reported to the Control Room serves as verification. A call to the National Earthquake Center will verify that an earthquake has occurred but will not provide ground acceleration data. Damage to some portions of the site may occur as a result of the felt earthquake but it should not affect the ability of safety functions to operate. This event escalates to an Alert under EAL HA4.1 if the earthquake adversely affects plant safety functions.

Reference(s):

1. HU1 – Natural and destructive phenomena affecting the Protected Area
2. AOP-28 Seismic Event
3. FSAR Volume 1 Section 2.9 Seismology
4. STPT 22.1 Seismic Event Monitoring
5. EPRI document, "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989

Category: Hazards **Sub-category:** Natural Events

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU4.2	**Unusual Event**
Sustained (15 min. average) winds ≥75 mph onsite **<u>OR</u>** Report by plant personnel of tornado striking within plant Protected Area	

Mode Applicability:

All

Basis:

This EAL is based on the assumption that a tornado striking (touching down) or hurricane force winds (≥75 mph) within the Protected Area may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL # HA4.2.

Reference(s):

1. NEI HU1 – Natural or destructive phenomena affecting the Protected Area
2. FSAR Volume 3 Page 5.1-37 Wind and Tornado Forces
3. PSA Section 9 Notebook 9.1

Category: Hazards **Sub-category:** Natural Events

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU4.3	**Unusual Event**
Uncontrolled flooding in the auxiliary building caused by rupture of the SW header **<u>OR</u>** Uncontrolled flooding in the water intake structure caused by rupture of a circulating water system expansion joint or fire water main.	

Mode Applicability:

All

Basis:

This EAL 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 via EAL # HA4.3.

Reference(s):

1. NEI HU1 – Natural or destructive phenomena affecting the Protected Area
2. PSA Section 7.3 Plant Flood Design Basis
3. PBNP Individual Plant Examination (IPE) for internal events and internal flood.

Category: Hazards **Sub-category:** Natural Events

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA4.1 **Alert**

Two or more seismic monitors (SEI 6210 through 6213) indicate ground acceleration **EITHER**:

≥0.06 g horizontal

OR

≥0.04 g vertical

Mode Applicability:

All

Basis:

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 safety systems.

This EAL 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 cause damage to plant structures, systems or equipment and therefore represent a potential substantial degradation of the plant.

Seismic monitors are located in the following areas:

SEI 6210 #3 Warehouse
SEI 6211 Unit 1 Facade
SEI 6212 Drum Prep. Room
SEI 6213 El 8' between vital switchgear room and AFW Tunnel

Reference(s):

1. NEI HA1 – Natural and destructive phenomena affecting the plant Vital Area
2. AOP-28 Seismic Event
3. FSAR Volume 1 Section 2.9 Seismology
4. STPT 22.1 Seismic Event Monitoring
5. EPRI document, Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989

Category: Hazards **Sub-category:** Natural Events

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA4.2 **Alert**

Sustained (15 minute average) winds ≥108 mph onsite

OR

Tornado strikes any Table H-1 Safe Shutdown Area,

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

Mode Applicability:

All

Basis:

Sustained 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. 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.

This EAL is based on the FSAR design basis sustained wind speed of 108 mph. Wind loads of this magnitude can cause damage to safety functions.

Reference(s):

1. NEI HA1 – Natural and destructive phenomena affecting the plant Vital Area
2. FSAR Volume 3 Page 5.1-37 Wind and Tornado Forces
3. PSA Section 9 Notebook 9.1

Category: Hazards **Sub-category:** Natural Events

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA4.3 **Alert**

Uncontrolled flooding that results in degraded safety system performance or that creates industrial safety hazards that precludes access necessary to operate or monitor safety equipment in **EITHER**:

The auxiliary building caused by rupture of the SW header

OR

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

Mode Applicability:

All

Basis:

This EAL addresses 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.

Reference(s):

1. NEI HA1 – Natural and destructive phenomena affecting the plant Vital Area
2. PSA Section 7 .3 Plant Flood Design Basis - Component Vulnerability and Table 7.7-1
3. PBNP Individual Plant Examination (IPE) for internal events and internal flood.

Category: Other **Sub-category:** N/A

Initiating Condition: Emergency Director Judgment

EAL:

HU6.1	**Unusual Event**
Any event, in the judgment of the Emergency Director, that could lead to or has led to a potential degradation of the level of safety of the plant.	

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event 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.

Reference(s):

1. NEI HU5 – Other conditions existing which in the judgment of the Emergency Director warrant declaration of an Unusual Event

Category: Other **Sub-category:**N/A

Initiating Condition: Emergency Director Judgment

EAL:

HA6.1	**Alert**
Any event in the judgment of the Emergency Director, that could cause or has caused actual substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of EPA Protective Action Guides.	

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

Reference(s):

1. NEI HA6 – Other conditions existing which in the judgment of the Emergency Director warrant declaration of an alert.
2. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents

Category: Other **Sub-category:** N/A

Initiating Condition: Emergency Director Judgment

EAL:

HS6.1	**Site Emergency**
Any event in the judgment of the Emergency Director is in progress which indicates actual or likely failures of plant systems needed to protect the public. Any releases are not expected to result in exposures which exceed EPA Protective Action Guides beyond the site boundary.	

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Emergency.

Reference(s):

1. NEI HS3 – Other conditions existing which in the judgment of the Emergency Director warrant declaration of Site Emergency
2. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents

Category: Other **Sub-category:** N/A

Initiating Condition: Emergency Director Judgment

EAL:

HG6.1	**General Emergency**
Any event in the judgment of the Emergency Director is in progress which indicates actual or imminent core damage and the potential for a large release of radioactive material in excess of EPA Protective Action Guides outside the site boundary.	

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.
Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the site boundary.

Reference(s):

1. NEI HG2 – Other conditions existing which in the judgment of the Emergency Director warrant declaration of General Emergency
2. EPA 400, Manual of Protective Action Guides And Protective Actions For Nuclear Incidents

Category: ISFSI Events **Sub-category:** Loss of Cask Confinement

Initiating Condition: Damage to a loaded cask confinement boundary

EAL:

IU1.1 **Unusual Event**

Loss of cask confinement boundary as indicated by exceeding any of the following external surface dose rates on any loaded Dry Storage Cask:

- $\geq$100 mR/hr at the cask side
- $\geq$200 mR/hr at the top of the cask
- $\geq$350 mR/hr at the cask air inlet
- $\geq$100 mR/hr at the cask air outlet

Mode Applicability:

All

Basis:

An Unusual Event in this EAL is declared on the basis of the occurrence of any event, natural or accident, 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.

The specified surface dose rates are based on the ISFSI Safety Analysis Report (SAR) design criteria referenced in the cask Certificate of Compliance and the related NRC Safety Evaluation Report. This EAL addresses responses to a dropped cask, a tipped over cask, explosion, missile damage, fire damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

Reference(s):

1. NEI E-HU1 – Damage to a loaded cask confinement boundary
2. Conditions for Cask Use and Technical Specifications Certificate of Compliance for the VSC-24 Dry Cask Storage System
3. Technical Specification 1.2.4 of the VSC-24 Certificate of Compliance
4. AOP-8G Dry Fuel Storage Cask Drop or Tipover
5. Conditions for Cask Use and Technical Specifications Certificate of Compliance for the NUHOMS® 32-PT Dry Cask Storage System
6. Technical Specification 1.2.7.a of the NUHOMS® 32-PT Certificate of Compliance

Category: ISFSI Events **Sub-category:** Security

Initiating Condition: Confirmed security event with potential loss of level of safety of the ISFSI

EAL:

IU1.2	**Unusual Event**
Report by Security Shift Supervisor of a security concern within the ISFSI	

Mode Applicability:

All

Basis:

This EAL is based on the PBNP Safeguards Contingency Plan. Security events that 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 Supervision because this individual is the designated on-site person 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 Safeguards Contingency Plan.

Reference(s):

1. NEI E-HU2 – Confirmed security event with potential loss of level of safety of the ISFSI

Category: ATWS **Sub-category:** N/A

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

EAL:

MA7.1 **Alert**

Any failure of the Reactor Protection System to generate an automatic trip signal and reduce power range to ≤ 5%

AND

Manual trip is successful

Mode Applicability:

1- Power Operation, 2-Startup

Basis:

This EAL indicates failure of the automatic protection system to trip the reactor. This condition is a potential substantial 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 (even if manual trip is successful) because conditions exist that lead to challenge of Fuel Cladding or Reactor Coolant System barrier integrity. Reactor Protection System trip setpoint signal being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because the automatic protection system is the issue. Following a successful reactor trip, nuclear power promptly drops to only a few percent of nominal, and then decays away to a level some 8 decades less. Reactor power levels resulting from radioactive fission product decay are never more than a few percent of nominal power and also lower in time. Heat removal safety systems are sized to remove only decay heat and not significant core power. Reactor power levels at or above 5% (in a core that is supposed to be shutdown) are considered an extreme challenge to the Fuel Cladding barrier and warrant a Critical Safety Function Status Tree (CSFST) RED priority. The setpoint has been chosen because it is clearly readable on the power range meters. Reactor power levels in the power range are indicated on N-41, 42, 43 and 44.
Following any automatic reactor trip signal, plant procedures prescribe operator insertion of redundant manual trip signals to ensure reactor shutdown is achieved. A successful manual trip is any set of actions by the reactor operator(s) at the Main Control Panel that causes control rods to be rapidly inserted into the core and brings the reactor subcritical. Failure of the manual trip would escalate the event to a Site Emergency (EAL# MS7.1).

Reference(s):

1. NEI 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
2. CSP-ST.0, Critical Safety Function Status Trees, Figure 1
3. BG-CSP-ST.0, Critical Safety Function Status Trees

Category: ATWS **Sub-category:** N/A

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

EAL:

MS7.1	**Site Emergency**
Conditions requiring entry into Critical Safety Function - Subcriticality-RED path (CSP-S.1)	

Mode Applicability:

1- Power Operation, 2-Startup

Basis:

Critical Safety Function Status Tree (CSFST) Subcriticality-RED path is entered based on failure of power range indication to lower below 5% following a reactor trip. Reactor power levels in the power range are indicated on N-41, 42, 43 and 44. This addresses any manual trip or automatic trip signal followed by a manual trip that fails to shut down the reactor to an extent that the reactor is producing more heat load for which the safety systems were designed. A manual trip is any set of actions by the reactor operator(s) at the main control board which causes control rods to be rapidly inserted into the core and brings power below that percent power (5%) associated with the ability of the safety systems to remove heat and continue to lower. Automatic and manual trips are not considered successful if action away from the main control board is 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. Emergency boration is thus required and there is an actual major failure of a system intended for protection of the public. The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Cladding and Reactor Coolant System barriers and warrants declaration of a Site Emergency.

Escalation of this event to a General Emergency would be via EAL# MG7.1 or other EAL categories.

Reference(s):

1. NEI SS2 – Failure of Reactor Protection System instrumentation to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was not successful
2. CSP-ST.0, Critical Safety Function Status Trees
3. BG-CSP-ST.0, Critical Safety Function Status Trees
4. CSP-S.1, Response to Nuclear Power Generation/ATWS

Category: ATWS **Sub-category:** N/A

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

EAL:

MG7.1	**General Emergency**

Conditions requiring entry into Subcriticality-RED path (CSP-S.1) currently exist

AND

Conditions requiring entry into **EITHER**:

Core Cooling-RED path (CSP-C.1)

OR

Heat Sink-RED path (CSP-H.1)

Mode Applicability:

1- Power Operation, 2-Startup

Basis:

Critical Safety Function Status Tree (CSFST) Subcriticality-RED path is entered based on failure of power range indication to lower below 5% following a reactor trip. Reactor power levels in the power range are indicated on N-41, 42, 43 and 44. This addresses any manual trip or automatic trip signal followed by a manual trip that fails to shut down the reactor to an extent that the reactor is producing more heat load for which the safety systems were designed. A manual trip is any set of actions by the reactor operator(s) at the main control board which causes control rods to be rapidly inserted into the core and brings power below that percent power (5%) associated with the ability of the safety systems to remove decay heat.

Core Cooling-RED path is entered if:

- Core exit thermocouples are equal to or greater than 1200°F, or
- Core exit thermocouples are less than 1200°F but equal to or greater than 700°F and all of the following:
  - RCS subcooling based on core exit thermocouples is equal to or less than [80°F] 35°F
  - No RCP is running
  - RVLIS NR equal to or less than 25 ft

Either set of conditions indicates significant core exit superheating and core uncovery. This is considered a loss of the Fuel Cladding barrier.

Heat Sink-RED path is entered if narrow range level in any steam generator (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. Heat Sink-RED therefore addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a challenge of the Fuel Cladding and RCS barriers.

The combination of these conditions (reactor power greater than 5% and either Core Cooling-RED path or Heat Sink-RED path) indicates the ultimate heat sink function is under extreme challenge. A major consideration is the inability to initially remove heat during the early stages of this core melt sequence.

In the event this challenge occurs at a time when the reactor has not been brought below the power associated with safety system design power (5%), a core melt sequence may exist and rapid degradation of the fuel cladding could begin. To permit maximum offsite intervention time, the General Emergency declaration is therefore appropriate in anticipation of an inevitable General Emergency declaration due to loss and challenge of fission product barriers.
CSFST setpoints enclosed in brackets (e.g., [51%], etc.) are used under adverse containment conditions.

Reference(s):

1. NEI 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
2. CSP-ST.0, Critical Safety Function Status Trees, Figures 1, 2 and 3
3. CSP-S.1, Response to Nuclear Power Generation/ATWS
4. CSP-C.1, Response to Inadequate Core Cooling
5. CSP-H.1, Response to Loss of Secondary Heat Sink

Category: Loss of AC Power Sources **Sub-category:** N/A

Initiating Condition: Loss of all offsite power to essential busses for ≥ 15 minutes

EAL:

MU8.1	**Unusual Event**
Unplanned loss of offsite AC power to both safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥15 minutes.	

Mode Applicability:

All

Basis:

Prolonged loss of all offsite 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 (station blackout).

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If neither of the unit safety-related buses are energized by an offsite source within 15 minutes, an Unusual Event is declared under this EAL. "Unplanned" loss of offsite power excludes scheduled maintenance and testing activities for which contingency plans have been established.

This EAL is the hot conditions equivalent of the cold conditions loss of offsite power EAL# MU8.1.

Reference(s):

1. NEI SU1 – Loss of all offsite power to essential busses for greater than 15 minutes
2. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
3. DBD-18, 13.8 KVAC System, Section 3.3.0
4. ECA 0.0
5. FSAR Section 8, Electrical Systems
6. AOP-14A

Category: Loss of AC Power Sources **Sub-category:** N/A

Initiating Condition: AC power capability to essential busses reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in station blackout

EAL:

MA8.1 **Alert**

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 ≥15 min. (one source away from station blackout):

- o A single emergency diesel generator (G01, G02, G03 or G04)
- o LVSAT 1(2)-X04
- o UAT 1(2)-X02
- o Cross-tying with the opposite unit power supply

Mode Applicability:
1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:
The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single failure would result in a station blackout. 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. The subsequent loss of the single remaining power source escalates the event to a Site Emergency (EAL# MS8.1).

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit safety-related buses within 15 minutes, an Alert is declared under this EAL.

"Unplanned" loss of offsite power excludes scheduled maintenance and testing activities for which contingency plans have been established.

Reference(s):

1. NEI 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
2. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
3. DBD-18, 13.8 KVAC System, Section 3.3.0
4. ECA 0.0
5. FSAR Section 8, Electrical Systems
6. AOP-14A

Category: Loss of AC Power Sources **Sub-category:** N/A

Initiating Condition: Loss of all offsite power and loss of all onsite AC power to essential busses

EAL:

MS8.1	**Site Emergency**
Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥15 min.	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Loss of all AC power compromises all plant safety systems requiring electrical power. This EAL is indicated by the loss of all offsite and onsite AC power to the safety-related 4160 VAC buses. Prolonged loss of all AC power will cause core uncovery and loss of containment integrity; thus, this event can escalate to a General Emergency (EAL# MG8.1). The fifteen-minute interval was selected as a threshold to exclude transient power losses.
This EAL is the hot conditions equivalent of the cold conditions loss of all AC power EAL # MA8.2.

Reference(s):

1. NEI SS1 – Loss of all offsite power and loss of all onsite AC power to essential busses
2. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
3. DBD-18, 13.8 KVAC System, Section 3.3.0
4. ECA 0.0
5. FSAR Section 8, Electrical Systems
6. AOP-14A

Category: Loss of AC Power Sources **Sub-category:** N/A

Initiating Condition: Prolonged loss of all offsite power and prolonged loss of all onsite AC power to essential busses

EAL:

MG8.1	**General Emergency**

Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06

AND EITHER:

Power restoration to any safety-related 4160 VAC bus or 480 VAC bus is not likely in $\leq$4 hours

OR

Conditions require entry into Core Cooling-RED path (CSP-C.1) or Core Cooling-ORANGE path (CSP-C.2)

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Loss of all AC power compromises all plant safety systems requiring electrical power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of Fuel Cladding, RCS and Containment barriers. The four-hour interval to restore AC power is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout." Although this EAL may be viewed as redundant to the Fission Product Barrier EALs, its inclusion is necessary to better assure timely recognition and emergency response.

The likelihood of restoring at least one safety-related 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 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 challenge of fission product barriers is imminent?
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 challenge 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 challenge of fission product barriers and degraded ability to monitor fission product barriers.
Either set of conditions indicates significant core exit superheating and core uncovery. This is considered a loss of the Fuel Cladding barrier.
Critical Safety Function Status Tree (CSFST) setpoints enclosed in brackets (e.g., [120 ft], etc.) are used under adverse containment conditions.

Reference(s):

1. NEI SG1 – Prolonged loss of all offsite power and prolonged loss of all onsite AC power to essential busses
2. DBD-T-46, Section 3.1
3. 10 CFR 50.63 and Regulatory Guide 1.155, Station Blackout
4. DBD-22, 4160 VAC System, Figure 1-1 & Section 5
5. DBD-18, 13.8 KVAC System, Section 3.3.0
6. ECA 0.0
7. FSAR Section 8, Electrical Systems
8. AOP-14A
9. CSP-C.1, Response to Inadequate Core Cooling
10. CSP-C.2, Response to Degraded Core Cooling

Category: Loss of DC Power Sources **Sub-category:** N/A

Initiating Condition: Loss of all vital DC power

EAL:

MS9.1	**Site Emergency**
≤105 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for ≥15 min. due to unplanned activities	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovery and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. 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 105 volts.

This EAL is the hot conditions equivalent of the cold conditions loss of DC power EAL# MU9.1.

Reference(s):

1. NEI SS3 – Loss of all vital DC power
2. FSAR Section 8.7
3. 0-SOP-DC-001/2/3/4 Section 3.8

Category: Equipment Failures **Sub-category:** Technical Specifications

Initiating Condition: Inability to reach required shutdown within Technical Specification limits

EAL:

MU10.1	**Unusual Event**
Plant is not brought to required operating mode within Technical Specifications LCO required action completion time.	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification 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 Technical Specification requires a one-hour report under 10CFR50.72 (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 declaration of an Unusual Event 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 an Unusual Event is based on the time at which the LCO-specified action completion period elapses under Technical Specifications and is not related to how long a condition may have existed. Other Technical Specification shutdowns that involve precursors to more serious events are addressed by other EALs.

Reference(s):

1. NEI SU2 – Inability to reach required shutdown within Technical Specification limits
2. Point Beach Nuclear Plant Technical Specifications

Category: Equipment Failures **Sub-category:** Turbine Failures

Initiating Condition: Natural or destructive phenomena affecting the Protected Area (turbine)

EAL:

MU11.1	**Unusual Event**
Report of main turbine failure requiring turbine trip resulting in: Damage to turbine-generator seals . **OR** Casing penetration	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

This EAL 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 significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this EAL to classify minor operational leakage. This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Reference(s):

1. NEI HU1 – Natural and destructive phenomena affecting the Protected Area
2. ARP 1(2) C33 1-2 Hydrogen Pressure High-Low Alarm
3. ARP 1(2) C33 1-3 Hydrogen Supply Pressure Low Alarm
4. AOP-5A Loss of Condenser Vacuum

Category: Equipment Failures **Sub-category:** Turbine Failures

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area (turbine)

EAL:

MA11.1	**Alert**
Turbine failure generated missiles resulting in visible damage to or penetrating any Table H-1 Safe Shutdown Area structure or system	

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

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

This EAL addresses the threat to safety-related equipment imposed by missiles generated by main turbine rotating component failures. This EAL is 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.

Reference(s):

1. HA1 – Natural or destructive phenomena affecting the plant Vital Area
2. PSA Section 8.0 Fire Hazards Analysis Table 8.1.3-1 Safe Shutdown Systems
3. PBNP FSAR Section 14.1.12 Likelihood of T-G Unit Overspeed
4. WSTG-4-NP, Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors, 1984

Category: Equipment Failures **Sub-category:** Loss of Indications/Alarms

Initiating Condition: Unplanned loss of most or all safety system annunciation or indication in the control room for ≥ 15 minutes

EAL:

MU12.1	**Unusual Event**
Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 minutes.	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment. The availability of computer based indication equipment is considered (i.e., PPCS).

"Unplanned" loss of annunciators or indicators excludes scheduled maintenance and testing activities. If safety system annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected.

Plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification. The fifteen-minute interval offers time to recover from transient or momentary power losses. If a transient is in progress during the loss of annunciation or indication, the event escalates to an Alert classification.

Reference(s):

1. NEI SU3 – Unplanned loss of most or all safety system annunciation or indication in the control room for greater than 15 minutes
2. FSAR Section 7.6
3. OM 1.1, Conduct of Plant Operations, Attachment 2

Category: Equipment Failures **Sub-category:** Loss of Indications/Alarms

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

EAL:

MA12.1 **Alert**

Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 min.

AND EITHER:

A significant transient is in progress

OR

PPCS is unavailable

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. The availability of computer based indication equipment is considered (i.e., PPCS).

"Unplanned" loss of annunciators or indicators does not include the audible feature of the annunciator or scheduled maintenance and testing activities.

If safety system annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions.

The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

"Significant transient" includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, or ECCS injections.

If the operating crew cannot monitor the transient in progress, the Alert escalates to a Site Emergency via MS12.1.

Reference(s):

1. NEI 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
2. FSAR Section 7.6
3. OM 1.1, Conduct of Plant Operations, Attachment 2

Category: Equipment Failures **Sub-category:** Loss of Indications/Alarms

Initiating Condition: Inability to monitor a significant transient in progress

EAL:

MS12.1 **Site Emergency**

Unplanned loss of annunciators or indicators on any 2 Control Room panels C01, C02, 1C03, 2C03, 1C04, 2C04, 1C20, or 2C20 for ≥15 min.

AND

PPCS is unavailable

AND

Complete loss of ability to monitor all critical safety function status

AND

A significant transient is in progress

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

This EAL recognizes the inability of the Control Room staff to monitor the plant response to a transient. A Site Emergency exists if the Control Room staff cannot monitor safety functions needed for protection of the public.

If safety system annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected.

"Significant transient" includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, or ECCS injections

Indications needed to monitor critical safety functions necessary for protection of the public must include Control Room indications, computer generated indications (PPCS) and dedicated annunciation capability. The specific indications should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, remove heat from the core, and maintain the reactor coolant system and containment intact.

Planned actions are included in the EAL since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Reference(s):

1. NEI SS6– Inability to monitor a significant transient in progress
2. FSAR Section 7.6
3. OM 1.1, Conduct of Plant Operations, Attachment 2

Category: Fission Product Barriers **Sub-category:** N/A

Initiating Condition: Any loss or challenge of Containment

EAL:

FU1.1	**Unusual Event**
Any loss or challenge of Containment (Table F-1, page 94).	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases, and references.

Fuel Cladding and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Fuel Cladding and RCS barriers, the loss of either of which results in an Alert (EAL# FA1.1), loss of the Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or challenge of the Containment barrier in combination with the loss or challenge of either the Fuel Cladding or RCS barrier results in declaration of a Site Emergency. (EAL# FS1.1).

Reference(s):

1. NEI FU1 - Table 5-F-4 – Any loss or challenge of containment

Category: Fission Product Barriers **Sub-category:**N/A

Initiating Condition: Any loss or any challenge of either Fuel Cladding or RCS

EAL:

FA1.1	**Alert**
Any loss or challenge of Fuel Cladding or RCS (Table F-1, page 92 and 93).	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases, and references.

At the Alert classification level, Fuel Cladding and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or challenge of either the Fuel Cladding or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or challenge of Containment barrier in combination with loss or challenge of either Fuel Cladding or RCS barrier results in declaration of a Site Emergency (EAL# FS1.1).

Reference(s):

1. NEI FA1 - Table 5-F-4 – Any loss or any challenge of either Fuel Cladding or RCS

Category: Fission Product Barriers **Sub-category:**N/A

Initiating Condition: Loss or challenge of any two barriers

EAL:

FS1.1	**Site Emergency**
Loss or challenge of any two barriers (Table F-1, pages 92, 93, and 94).	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases, and references.
At the Site Emergency classification level, each barrier is weighted equally.

Reference(s):

1. FS1 - Table 5-F-4 – Loss or challenge of any two barriers
2. SS4– Loss of heat removal capability

Category: Fission Product Barriers **Sub-category:**N/A

Initiating Condition: Loss of any two barriers with loss or challenge of a third

EAL:

FG1.1	**General Emergency**
Loss of any two barriers with a loss or challenge of a third (Table F-1, pages 92, 93 and 94).	

Mode Applicability:

1- Power Operation, 2-Startup, 3-Hot Standby, 4-Hot Shutdown

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases, and references.

Reference(s):

1. NEI FG1 - Table 5-F-4 – Loss of any two barriers with loss or challenge of a third

Category: RCS **Sub-category:** RCS Temperature

Initiating Condition: Unplanned loss of decay heat removal capability with irradiated fuel in the reactor vessel

EAL:

MU13.1 **Unusual Event**

An unplanned event results in RCS temperature ≥200°F

OR

Loss of all RCS temperature and Reactor Vessel level indication for ≥15 min.

Mode Applicability:

5- Cold Shutdown, 6-Refuel

Basis:

This EAL is an Unusual Event 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.

Reference(s):

1. NEI CU4 – Unplanned loss of decay heat removal capability with irradiated fuel in the reactor vessel
2. Tech Specs Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
3. DBD-9, Reactor Coolant System, Sections 3.23.1 to 3.23.4, 3.24.1, 3.24.2, 3.25.1 and 3.25.2
4. OI 105, RECS Heatup/Cooldown Plotting
5. OP4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
6. DBD-27
7. DBD-T-44

Category: RCS **Sub-category:** RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown with irradiated fuel in the reactor vessel

EAL:

MA13.1	**Alert**
An unplanned event results in RCS temperature exceeding 200°F for ≥ Table M-1 duration*	

*If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, this EAL is not applicable.

Table M-1 RCS Reheat Duration Thresholds

Containment and RCS Barrier Status	Duration
RCS intact	60 min.*
Containment closure established **AND EITHER**: RCS not intact **OR** RCS reduced inventory	20 min.*
Containment closure not established **AND** RCS not intact	0 min.

Mode Applicability:

5- Cold Shutdown, 6-Refuel

Basis:

This EAL is 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 in which 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.

The first threshold in Table M-1 addresses complete loss of functions required for core cooling for greater than sixty minutes during Refuel and Cold Shutdown modes when RCS integrity is established (irrespective of the status of Containment Closure). As in the second and third thresholds, 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 threshold is immaterial given that the RCS is providing a high-pressure barrier to fission product release to the environment. The sixty-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety.

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.
Containment closure should not be confused with refueling containment integrity as defined in technical specifications.
The second threshold in Table M-1 addresses the complete loss of functions required for core cooling for greater than twenty minutes during Refuel 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 the third threshold, 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 twenty-minute interval is 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. The asterisk highlights the note at the top of the table. The note indicates that the second threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the twenty-minute interval.
The third threshold in Table M-1 addresses complete loss of functions required for core cooling during Refuel 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 the third condition because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.
Escalation to a Site Emergency would be via EAL # MS15.1, MS15.2 or MS15.3 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

Reference(s):

1. NEI CA4 – Inability to maintain plant in cold shutdown with irradiated fuel in the reactor vessel
2. Tech Specs Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
3. Tech Specs 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

Category: RCS **Sub-category:** RCS Pressure

Initiating Condition: Inability to maintain plant in cold shutdown with irradiated fuel in the reactor vessel

EAL:

MA14.1	**Alert**
Unplanned RCS pressure rise ≥10 psig due to loss of decay heat removal	

Mode Applicability:

5- Cold Shutdown, 6-Refuel

Basis:

This EAL is not applicable during solid plant conditions. The pressure rise of 10 psig infers an RCS temperature in excess of the Technical Specification cold shutdown limit (200°F) for which EAL# MA13.1 would permit up to sixty minutes to restore RCS cooling before declaration of an Alert. This EAL therefore covers situations in which it is determined that, due to high decay heat loads, the time provided to reestablish temperature control should be less than sixty minutes.

This EAL is 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 in which 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.

Escalation to a Site Emergency would be via EAL # MS15.1, MS15.2 or MS15.3 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

Reference(s):

1. NEI CA4 – Inability to maintain plant in cold shutdown with irradiated fuel in the reactor vessel
2. Tech Specs Table 1.1-1, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
3. OP-1A, Cold Shutdown to Hot Standby, Step 5.3.2

Category: RCS **Sub-category:** RCS Level

Initiating Condition: Unplanned loss of RCS inventory with irradiated fuel in the Reactor Vessel

EAL:

MU15.1	**Unusual Event**
Unplanned RCS level lowering below 77.1% (1 foot below RPV flange) for ≥15 min. **OR** If Reactor Vessel level cannot be monitored, loss of RPV inventory as indicated by unexplained Containment Sump A level rise	

Mode Applicability:

6-Refuel

Basis:

This EAL is an Unusual Event 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 operations that lower RCS water level significantly below the Reactor Vessel flange are carefully planned and procedurally controlled. An unplanned event that results in water level lowering 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 fifteen-minute interval 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, a more serious condition may exist.

The first condition involves a lowering in RCS level below normal that continues for fifteen minutes due to an unplanned event. This EAL is not applicable to drops in flooded reactor cavity level (covered by lowering spent fuel pool water level in EAL# MU4.1) until such time as the level lowers to the level of the vessel flange. If level continues to lower and reaches the bottom inside diameter of the RCS loop (33 ft 2-7/8 in. elev. or 0%/0 in.), escalation to the Alert level via EAL# MA15.1 would be appropriate. If the level lowering is accompanied by RCS heatup, escalation to the Alert level via EAL# MA13.1 may also be appropriate.

In the second condition of this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by sump level changes. OI 55 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting. Containment Sump A is equipped with a high level alarm (80%). Sump 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.

Reference(s):

1. NEI CU2 – Unplanned loss of RCS inventory with irradiated fuel in the reactor vessel
2. OP4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
3. OI-55, Primary Leak Rate Calculation
4. OM 3.19, Reactor Coolant System Leakage Determination
5. DBD-33, Containment Structures And Penetrations Design Basis Document
6. ARB C01 B 1-4, Unit 1 Containment Sump A Level High
7. STPT 12.1, Waste Disposal System, R2v. 8
8. FSAR Section 7.6

Category: RCS **Sub-category:** RCS Level

Initiating Condition: Loss of reactor vessel inventory with irradiated fuel in the Reactor Vessel

EAL:

MA15.1 **Alert**

Loss of RCS or Reactor Vessel inventory as indicated by **EITHER**:

LI-447 and LI-447A $\leq$ 0% when aligned

OR

If RCS or Reactor Vessel level cannot be monitored for $\geq$15 min., loss of inventory as indicated by unexplained Containment Sump A level rise

Mode Applicability:

5-Cold Shutdown, 6-Refuel

Basis:

This EAL serves as a precursor 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 RCS or Reactor Vessel level lowering and potential core uncovery. The LI-447 and LI-447A threshold corresponds to the bottom inside diameter of the RCS loop. The bottom inside diameter of the RCS loop is the level equal to the bottom of the Reactor Vessel loop penetration, not the low point of the loop. This level was chosen because remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

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 level changes. OI 55 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting.

Sump 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 interval for the loss of level indication was chosen because it is half of the Site Emergency EAL duration. The interval allows this EAL to be an effective precursor to the Site Emergency EAL# MS15.2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than one hour. Therefore this EAL meets the definition for an Alert emergency.

Reference(s):

1. NEI CA1, CA2 – Loss of reactor vessel inventory with irradiated fuel in the reactor vessel
2. OP 4D Part 3, Reactor Cavity and Reactor Coolant System, Table 1
3. OI-55, Primary Leak Rate Calculation
4. OM 3.19, Reactor Coolant System Leakage Determination
5. Volian Enterprises Calculation WEP-STP-25
6. PBNP FSAR 4.0 Reactor Coolant System Design Basis, Table 4.1-6

Category: RCS **Sub-category:** RCS Level

Initiating Condition: Loss of reactor vessel inventory affecting core decay heat removal capability with irradiated fuel in the Reactor Vessel

EAL:

MS15.1	**Site Emergency**
With containment closure not established, RVLIS NR ≤[33 ft] 30 ft	
OR	
With containment closure established , RVLIS NR ≤[30 ft] 27 ft	

Mode Applicability:

5-Cold Shutdown, 6-Refuel

Basis:

Under the conditions specified by this EAL, continued lowering in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The level associated without containment closure corresponds to six inches below the bottom inside diameter of the RCS loop. The level associated with containment closure not established corresponds to the top of active fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel level lowering and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a loss of the RCS barrier and challenge of the Fuel Cladding barrier.

- Setpoints enclosed in brackets (e.g., [33 ft], etc.) are used under adverse containment conditions.

Containment closure is the action taken to secure containment and its assorted structures, systems and components as a functional barrier to fission product release under existing plant conditions.
Containment closure is initiated per the SEPs or Shift Manager direction if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires that, upon a loss of decay heat removal, any open penetration which is listed on CL 1E, Containment Closure Checklist, must be closed or capable of being closed prior to RCS bulk boiling.
This checklist is maintained any time that the RCS is <200°F and containment operability is not maintained.
Containment closure should not be confused with refueling containment integrity as defined in technical specifications.

Reference(s):

1. NEI CS1, CS2 – Loss of reactor vessel inventory affecting core decay heat removal capability with irradiated fuel in the reactor vessel
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
5. Tech Specs B 3.6.1, Containment, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
6. CL 1E, Containment Closure Checklist

Category: RCS **Sub-category:** RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting core decay heat removal capability

EAL:

MS15.2	**Site Emergency**
Reactor Vessel level cannot be monitored for ≥30 min. **AND** A loss of Reactor Vessel inventory as indicated by **EITHER**: Unexplained Containment Sump A level rise **OR** Erratic Source Range Monitor indication	

Mode Applicability:

5-Cold Shutdown

Basis:

Under the conditions specified by this EAL, continued lowering in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel.

Declaration is therefore associated simply with the lowering inventory trend rather than indications of actual core uncovery. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling.

In the Cold Shutdown mode, normal RCS level indication (e.g., RVLIS) may be unavailable and, the Reactor Vessel inventory loss must be detected by sump level changes. OI 55 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting.

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

Analysis indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

The thirty-minute interval 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 EAL# MA15.1 and MS15.1. Escalation to a General Emergency is via EAL# MG15.1.

Reference(s):

1. NEI CS1 – Loss of reactor vessel inventory affecting core decay heat removal capability
2. OI-55, Primary Leak Rate Calculation
3. OM 3.19, Reactor Coolant System Leakage Determination
4. OP 1B, Reactor Startup
5. Generic Letter 88-17, Loss of Decay Heat Removal
6. SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues
7. NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States
8. NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Category: RCS **Sub-category:** RCS Level

Initiating Condition: Loss of reactor vessel inventory affecting core decay heat removal capability with irradiated fuel in the Reactor Vessel

EAL:

MS15.3	**Site Emergency**
Reactor Vessel level cannot be monitored **<u>AND</u>** Indication of core uncovery as evidenced by one or more of the following: o Containment High Range Radiation Monitor reading $\geq$10 R/hr o Erratic Source Range Monitor indication o Unexplained Containment Sump A level increase	

Mode Applicability:

6-Refuel

Basis:

Under the conditions specified by this EAL, continued lowering in Reactor Vessel level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach or continued boiling in the Reactor Vessel.

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, the thirty-minute interval was conservatively chosen.

In Refuel mode, normal RCS level indication (e.g., RVLIS) may be unavailable but alternate means of level indication are normally installed (including visual observation) 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 on-scale Containment High Range Monitor indication and possible alarm. The 10 R/hr setpoint has been selected to be well above that expected under normal plant conditions.
- 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.
- Sump level changes may be indicative of a loss of RCS inventory. Sump 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 effluent release is not expected with containment closure established; thus, declaration of a Site Emergency is warranted under the EAL conditions specified. Escalation to a General Emergency is via EAL# MG15.1.

Reference(s):

1. NEI CS2 – Loss of reactor vessel inventory affecting core decay heat removal capability with irradiated fuel in the reactor vessel
2. Eng Eval 2001-28, Containment High Radiation Channel Check Tolerance, 10/5/01
3. EPIP 10.2, Core Damage Estimation, Section 4.1
4. OI-55, Primary Leak Rate Calculation
5. OM 3.19, Reactor Coolant System Leakage Determination
6. OP 1B, Reactor Startup, Step 5.1
7. Generic Letter 88-17, Loss of Decay Heat Removal
8. SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues
9. NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States
10. NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Category: RCS **Sub-category:** RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting fuel cladding integrity with containment challenged and irradiated fuel in the Reactor Vessel

EAL:

MG15.1 **General Emergency**

1. Core uncovery for ≥30 min.

    as indicated by EITHER of the following:

    RVLIS NR ≤[30 ft] 27 ft

    OR

    One or more of the following when Reactor Vessel water level cannot be monitored:

    - Containment High Range Radiation Monitor reading ≥10 R/hr
    - Erratic Source Range Monitor indication
    - Unexplained Containment Sump A level rise

AND

2. Containment challenged as indicated by one or more of the following:
    - o Containment closure not established
    - o Hydrogen concentration in containment ≥6%
    - o Containment pressure ≥60 psig

Mode Applicability:

5-Cold Shutdown, 6-Refuel

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 core submergence cannot be restored as available decay heat will cause boiling and further lowers the vessel level.

Setpoints enclosed in brackets (e.g., [30 ft], etc.) are used under adverse containment conditions.

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, etc.) can have a significant impact on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.
If all means of level monitoring are not available, 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 on-scale Containment High Range Monitor indication and possible alarm. The 10 R/hr setpoint has been selected to be well above that expected under normal plant conditions.
- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered.
- Sump level changes may be indicative of a loss of RCS inventory. OI 55 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting. Sump 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.

Three conditions are associated a challenge to containment integrity:

- 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. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred.
- The containment design pressure (60 psig) is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and fuel clad boundaries in that it is not possible to reach this condition without severe core degradation.
- 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.

Containment closure should not be confused with refueling containment integrity as defined in technical specifications.

The General Emergency is declared on the occurrence of the loss or challenge of function of all three fission product barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause fuel cladding failure. With the containment breached or challenged, the potential for unmonitored fission product release to the environment is high. This is consistent with the definition of a General Emergency.

Reference(s):

1. NEI CG1 – Loss of reactor vessel inventory affecting fuel clad integrity with containment challenged with irradiated fuel in the reactor vessel
2. Eng Eval 2001-28, Containment High Radiation Channel Check Tolerance, 10/5/01
3. EPIP 10.2, Core Damage Estimation, Section 4.1
4. OI-55, Primary Leak Rate Calculation
5. OM 3.19, Reactor Coolant System Leakage Determination
6. OP 1B, Reactor Startup, Step 5.1
7. Volian Enterprises Calculation WEP-SPT-25
8. Tech Specs B 3.6.1, Containment, Unit 1 – Amendment No. 201/Unit 2 – Amendment No. 206
9. CSP-C.1 Unit 1 Red, Critical Safety Procedure Safety Related Response To Inadequate Core Cooling, Step 11
10. BG-CSP-Z.1, Response to High Containment Pressure, Step 11
11. EPIP 10.3, Post-Accident Containment Hydrogen Reduction
12. FSAR pg 5.1.35
13. BG-CSP-ST.0, CSFST, Step F.0.5
14. CL 1E, Containment Closure Checklist

Category: Loss of AC Power Sources **Sub-category:** N/A

Initiating Condition: Loss of all offsite power and loss of all onsite ac power to essential busses

EAL:

MA8.2	**Alert**
Loss of all AC power to safety-related 4160 VAC buses 1(2)-A05 and 1(2)-A06 for ≥15 min.	

Mode Applicability:

5-Cold Shutdown, 6-Refuel, D-Defueled

Basis:

Prolonged loss of all AC power compromises all plant safety systems requiring electrical power. This EAL is indicated by the loss of all offsite and onsite AC power to the safety-related 4160 VAC buses. The fifteen-minute interval was selected as a threshold to exclude transient power losses and to provide time to restore power prior to declaration.
This EAL is the cold conditions equivalent of the hot conditions loss of all AC power EAL # MS8.1.

Reference(s):

1. NEI CA3 – Loss of all offsite power and loss of all onsite ac power to essential busses
2. DBD-22, 4160 VAC System Figure 1-1 & Section 5
3. DBD-18, 13.8 KVAC System, Section 3.3.0
4. ECA 0.0
5. FSAR Section 8, Electrical Systems
6. AOP-14A

Category: Loss of DC Power Sources **Sub-category:** N/A

Initiating Condition: Unplanned loss of required DC power for greater than 15 minutes

EAL:

MU9.1	**Unusual Event**
≤105 VDC on 125 VDC buses D-01, D-02, D-03 and D-04 for ≥15 min. due to unplanned activities	

Mode Applicability:

5-Cold Shutdown, 6-Refuel

Basis:

The purpose of this 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.

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

Loss of DC power to any AC bus creates the following conditions:

- Associated breakers cannot be electrically opened or closed remotely or locally.
- Electrical protection/interlock tripping of associated breakers is rendered inoperable including undervoltage stripping. The one exception is the 480 V individual breaker overloads which remain operable.
- All associated breaker positions remain as-is.

This EAL is the cold conditions equivalent of the hot conditions loss of DC power EAL# MS9.1.

Reference(s):

1. NEI CU7 – Unplanned loss of required DC power for greater than 15 minutes
2. FSAR Section 8.7
3. 0-SOP-DC-001/2/3/4 Section 3.8

Table F-1
Fuel Cladding

Loss	Challenge
1. Conditions requiring entry into Core Cooling-RED path (CSP-C.1)	1. Conditions requiring entry into Core Cooling-ORANGE path (CSP-C.2)
2. Coolant activity ≥300 μCi/gm I-131 equivalent	2. Conditions requiring entry into Heat Sink-RED path (CSP-H.1)
3. CET readings ≥1200°F (Core Cooling-RED path, CSP-C.1)	3. CET readings ≥700°F
4. Containment rad monitor reading ≥17 R/hr	4. RVLIS NR ≤25 ft with no RCPs running
5. Failed Fuel Monitor (RE-109) reading ≥120 mRem/hr	5. Emergency Director Judgment
6. Emergency Director Judgment	

Table F-1
RCS

Loss	Challenge
1. RCS subcooling based on core exit thermocouples ≤[80°F] 35°F due to RCS leakage 2. SGTR in excess of available charging pumps 3. Containment rad monitor reading ≥3.0 R/hr 4. Emergency Director Judgment	1. Conditions requiring entry into RCS Integrity-RED path (CSP-P.1) 2. Conditions requiring entry into Heat Sink-RED path (CSP-H.1) 3. Unisolable leak exceeding 60 gpm 4. Emergency Director Judgment

Table F-1
Containment

Loss	Challenge
1. Rapid unexplained containment pressure drop following initial rise 2. Containment pressure or sump level response not consistent with LOCA conditions 3. Ruptured S/G is also faulted outside of containment 4. Primary-to-secondary leakage ≥10 gpm with non-isolable steam release from affected S/G to the environment 5. Containment isolation required and containment isolation or ventilation valve(s) not closed when required **AND** Radiological release pathway to the environment exists 6. Inability to isolate any primary system discharging outside containment **AND** Radiological release pathway to the environment exists 7. Emergency Director Judgment	1. Conditions requiring entry into Containment-RED path (CSP-Z.1) 2. Containment pressure ≥60 psig and rising (Containment-RED path, CSP-Z.1) 3. Hydrogen concentration in containment ≥6% 4. Containment pressure ≥25 psig with less than one train of containment spray and two containment accident fan cooler units operating 5. CET readings ≥1200°F (Core Cooling-RED path, CSP-C.1) **AND** Restoration procedures not effective within 15 min. 6. CET readings ≥700°F with RVLIS NR <25 ft and no RCPs running (Core Cooling-RED path, CSP-C.1) **AND** Restoration procedures not effective within 15 min. 7. Containment radiation ≥15,900 R/hr 8. Emergency Director Judgment

Bases

Fuel Cladding Challenge

1. **Conditions requiring entry into Core Cooling-ORANGE path (CSP-C.2)**

Core Cooling-ORANGE path is entered if Core Exit Thermocouples are reading less than 1200°F and RCS subcooling based on core exit thermocouples is equal to or less than [80°F] 35°F and any of the following:

- With two RCPs running, RVLIS WR is equal to or less than [120 ft] 110 ft
- With one RCP running, RVLIS WR is equal to or less than [100 ft] 90 ft
- With no RCP running either:
  - Core exit thermocouples are equal to or greater than 700°F and RVLIS NR is greater than 25 ft
  - Core exit thermocouples are less than 700°F and RVLIS NR is equal to or less than 25 ft

Any or these conditions indicate subcooling has been lost and that some fuel cladding damage may potentially occur. Critical Safety Function Status Tree (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 1E5 R or unknown.

Reference(s):

1. NEI Fuel Clad Challenge1 – Critical Safety Function Status Tree (CSFST) Core Cooling-Orange OR Heat Sink-Red
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. CSP-C.2, Response to Degraded Core Cooling

2. **Conditions requiring entry into Heat Sink-RED path (CSP-H.1)**

Heat Sink-Red path is entered if narrow range level in any steam generator (S/G) is equal to or less than [51%] 29% and total feedwater flow to both 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 challenge of the Fuel Cladding barrier. Critical Safety Function Status Tree (CSFST) setpoints enclosed in brackets (e.g., [51%], etc.) are used under adverse containment conditions. Adverse containment conditions are defined as either:

- 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 1E5 R or unknown.

Reference(s):

1. NEI Fuel Clad Challenge1 – CSFST Core Cooling-Orange OR Heat Sink-Red
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. CSP-H.1, Response to Loss of Secondary Heat Sink

3. **CET readings ≥700°F**

Core exit thermocouple (CET) readings are included in addition to the Critical Safety Functions (CSFs) to include conditions when the CSFs may not be in use (initiation after SI is blocked). CET readings greater than 700°F corresponds to the temperature reading for Core Cooling-ORANGE path in Fuel Cladding barrier Challenge #1. This temperature indicates subcooling has been lost and that some *cladding* damage may occur.

Reference(s):

1. NEI Fuel Clad Challenge 3 – Core Exit Thermocouple Readings GREATER THAN (site-specific) degree F
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2

4. **RVLIS NR ≤ 25 ft with no RCPs running**

RVLIS narrow range equal to or less than 25 ft with no RCPs running corresponds to a collapsed liquid level 3.5 feet above the bottom of the active fuel with core exit temperature greater than 700°F, including allowance for normal channel accuracy. This water level is an indication of inadequate coolant inventory and is used in the Core Cooling-ORANGE path and indicates subcooling has been lost and that some fuel cladding damage may occur.

Reference(s):

1. NEI Fuel Clad Challenge4 – Reactor Vessel Water Level LESS than (site-specific) value
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. Volian Enterprises Calculation No. WEP-SPT-25

5. **Emergency Director Judgment**

Emergency Director judgment addresses any other factors that are to be used in determining whether the fuel cladding is potentially lost. In addition, the inability to monitor the fuel cladding integrity should also be considered in this threshold as a factor in judging that the fuel cladding may be considered challenged.

Reference(s):

1. NEI Fuel Clad Challenge 7 – Emergency Director Judgment

Fuel Cladding Loss

1. **Conditions requiring entry into Core Cooling-RED path (CSP-C.1)**

Core Cooling-RED path is entered if:

- Core exit thermocouples are equal to or greater than 1200°F, or
- core exit thermocouples are less than 700°F and all of the following:
  - RCS subcooling based on core exit thermocouples is equal to or less than [80°F] 35°F
  - No RCP is running
  - RVLIS NR equal to or less than 25 ft

Either set of conditions indicates significant core exit superheating and core uncovery. This is considered a loss of the Fuel Cladding barrier. Critical Safety Function Status Tree (CSFST) setpoints enclosed in brackets (e.g., [80°F], 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 1E5 R or unknown.

Reference(s):

1. NEI Fuel Clad Loss 1 – CSFST Core Cooling-Red
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. CSP-C.1, Response to Inadequate Core Cooling

2. **Coolant activity ≥300 μCi/gm I-131 equivalent**

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. 300 μCi/gm Dose Equivalent I-131 is well above that expected for iodine spikes and corresponds to about 2% to 5% fuel cladding damage. When reactor coolant activity reaches this level, significant cladding heating has occurred and thus the Fuel Cladding barrier is considered lost.

Reference(s):

1. NEI Fuel Clad Loss 2 – Primary Coolant Activity GREATER THAN (site-specific) Value
2. NEI 99-01, Revision 4, pg 5-F-4

3. **CET readings ≥1200°F (Core Cooling-RED path, CSP-C.1)**

Core exit Thermocouple (CET) readings equal to or greater than 1200°F indicate significant core exit superheating and core uncovery. This is considered a loss of the Fuel Cladding barrier.

Reference(s):

1. NEI Fuel Clad Challenge3 – Core Exit Thermocouple Readings GREATER THAN (site-specific) degree F
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. CSP-C.1, Response to Inadequate Core Cooling

4. **Containment rad monitor reading ≥ 17 R/hr**

A containment radiation monitor reading greater than 17 R/hr is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/cc dose equivalent I-131 into the containment atmosphere. 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 (approximately 2-5 % cladding failure depending on core inventory and RCS volume). This value is higher than that specified for RCS barrier Loss #3.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping. 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

Reference(s):

1. NEI Fuel Clad Loss 5 – Containment rad monitor reading GREATER THAN (site-specific) R/hr
2. PBF 1608, Calculation 2004-0006
3. SAMG SAG-5, Reduce Fission Product Releases, Attachment D

5. **Failed Fuel Monitor (RE-109) reading ≥120 mRem/hr**

Fuel cladding damage in the range of 2% - 5% is generally considered the threshold for the loss of the Fuel Cladding barrier. Calc 96-0073 indicates 2,400 mRem/hr on 1(2) RE-106 corresponds to 100% fuel cladding damage. Five percent fuel cladding damage is therefore one-twentieth of the one hundred percent value or 120 mRem/hr.

Reference(s):

1. NEI Fuel Clad Loss 6 – Other (Site-Specific) Indications of fuel clad barrier loss
2. Calc 96-0073, 2/29/96, (NEPG-86-515)

6. **Emergency Director Judgment**

Emergency Director judgment addresses any other factors that are to be used in determining whether the fuel cladding is lost. In addition, the inability to monitor the fuel cladding integrity should also be considered in this threshold as a factor in judging that the fuel cladding may be considered lost.

Reference(s):

1. NEI Fuel Clad Challenge7 – Emergency Director Judgment

RCS Challenge

1. **Conditions requiring entry into RCS Integrity-RED path (CSP-P.1)**

RCS Integrity-Red path is entered if:
- Temperature drop in both cold legs is equal to or greater than 100°F, and
- Temperatures in both cold legs are equal to or less than 285°F.

The combination of these two conditions indicates the RCS barrier is under significant challenge and should be considered a challenge of RCS barrier.

Reference(s):

1. NEI RCS-Challenge 1 – CSFST RCS Integrity-Red OR Heat Sink-Red
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 4
3. CSP-P.1, Response to Imminent Pressurized Thermal Shock Condition

2. **Conditions requiring entry into Heat Sink-RED path (CSP-H.1)**

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 challenge of the RCS barrier.

Critical Safety Function Status Tree (CSFST) setpoints enclosed in brackets (e.g., [51%], 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 1E5 R or unknown.

Reference(s):

1. NEI RCS-Challenge 1 – CSFST RCS Integrity-Red OR Heat Sink-Red
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 3

3. **Unisolable leak exceeding 60 gpm**

This threshold is based on the inability to maintain normal liquid inventory within the RCS by normal operation of the Chemical and Volume Control System, which is considered as one centrifugal charging pump discharging to the charging header. The need for a second charging pump would be indicative of a substantial RCS leak. 60 gpm is the minimum operability flow rate for each charging pump.

Reference(s):

1. NEI RCS-Challenge 2 – RCS Leak Rate: Unisolable leak exceeding the capacity of one charging pump in the normal charging mode
2. DBD-04, Chemical and Volume Control System, Section 3.9

4. **Emergency Director Judgment**

The Emergency Director Judgment RCS challenge addresses any event-specific factors that may be indicative of a challenge to the RCS barrier. The inability to monitor RCS integrity should also be considered as a factor in judging that the RCS barrier may be considered potentially lost.

Reference(s):

1. NEI RCS-Challenge 6 – Emergency Director Judgment

RCS Loss

1. **RCS subcooling based on core exit thermocouples ≤[80°F] 35°F due to RCS leakage**

This threshold addresses conditions in which 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.

Reference(s):

1. NEI RCS-Loss 2 – RCS Leak Rate GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling
2. CSP-ST.0, Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. BG-CSP-ST.0 Step ST-2

2. **SGTR in excess of available charging pumps**

In conjunction with Containment barrier Loss #3 and the Fuel Cladding barrier thresholds, this threshold is intended to address the full spectrum of Steam Generator Tube Rupture (SGTR) events. To meet this threshold, the leakage must be 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.

Reference(s):

1. NEI RCS-Loss 3 – Steam generator tube rupture that results in an ECCS (SI) Actuation
2. EOP-0, REACTOR TRIP OR SAFETY INJECTION
3. DBD-04, Chemical and Volume Control System

3. **Containment rad monitor reading ≥3.0 R/hr**

The containment radiation monitor reading is a value that 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. The reading is less than that specified for Fuel Cladding barrier Loss #4 because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping. 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

Reference(s):

1. NEI RCS-Loss 4 – Containment rad monitor reading GREATER THAN (site-specific) R/hr
2. PBF 1608, Calculation 2004-0006
3. SAMG SAG-5, Reduce Fission Product Releases, Attachment D

4. **Emergency Director Judgment**

Emergency Director judgment addresses any other factors that are to be used in determining whether the RCS is lost. In addition, the inability to monitor the RCS integrity should also be considered in this threshold as a factor in judging that the RCS may be considered lost.

Reference(s):

1. NEI RCS-Loss 6 – Emergency Director Judgment

Containment Challenge

1. **Conditions requiring entry into Containment-RED path (CSP-Z.1)**

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. This threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation (metal-water reaction) or failure to trip in combination with RCS breach. This combination of conditions would be expected to require the declaration of a General Emergency.

Reference(s):

1. NEI Containment- Challenge 1 – CSFST Containment-Red
2. CSP-ST.0, Unit 1(2) Critical Safety Function Status Trees, Figure 5
3. BG-CSP-ST.0 Step ST-5
4. CSP-Z.1, Response to High Containment Pressure

2. **Containment pressure ≥60 psig and rising (Containment-RED path, CSP-Z.1)**

This threshold is the containment design pressure and is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation (metal-water reaction) or failure to trip in combination with RCS breach. This condition would be expected to require the declaration of a General Emergency.

Reference(s):

1. NEI Containment- Challenge 2 – Containment Pressure: (Site-specific) PSIG and increasing
2. FSAR pg 5.1.35
3. BG-CSP-ST.0, CSFST, Step F.0.5
4. CSP-Z.1, Response to High Containment Pressure

3. **Hydrogen concentration in containment ≥6%**

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. Since this threshold is also indicative of loss of both Fuel Cladding and RCS barriers with the challenge of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

Reference(s):

1. NEI Containment- Challenge 2 – Containment Pressure: Explosive mixture exists
2. CSP-C.1 UNIT 1 RED, CRITICAL SAFETY PROCEDURE SAFETY RELATED RESPONSE TO INADEQUATE CORE COOLING, Step 11
3. BG-CSP-Z.1, Response to High Containment Pressure, Step 11
4. EPIP 10.3, POST-ACCIDENT CONTAINMENT HYDROGEN REDUCTION

4. **Containment pressure ≥25 psig with less than one train of containment spray and two containment accident fan cooler units operating**

This threshold represents a challenge of containment in that the containment heat removal/depressurization equipment (but not including containment venting strategies) is either lost or performing in a degraded manner. One train of containment spray and two containment accident fan cooler units is defined to be one full train of depressurization equipment . This equipment will provide 100% of the required cooling capacity during post-accident conditions. Each containment spray system consists of a spray pump, spray header, nozzles, valves, piping, instruments, and controls to ensure an operable flow path capable of taking suction from the RWST upon an ESF actuation signal. Each containment accident fan cooler unit consists of cooling coils, accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path. The containment pressure setpoint (25 psig) is the pressure at which the equipment should have actuated and began performing its function.

Reference(s):

1. NEI Containment- Challenge 2 – Containment Pressure: Pressure greater than containment depressurization actuation setpoint with less than one full train of depressurization equipment operating
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 5
3. BG-CSP-ST.0 Step ST-5
4. TS B 3.6.6, pgs B 3.6.6-4 & -5, 10/20/02

5. **CET readings ≥1200°F (Core Cooling-RED path, CSP-C.1) AND Restoration procedures not effective within 15 min.**

This threshold indicates significant core exit superheating and core uncovery and is considered a loss of the Fuel Cladding barrier. It must also be assumed that the loss of RCS inventory is a result of a loss of the RCS barrier. These conditions, if not mitigated, will likely lead to core melt which will in turn result in a challenge of containment.

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 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 not been, or will not be effective.

For the purpose of this threshold the term 'effective' with regards to functional restoration procedures means that the specified criterion no longer exists.

Reference(s):

1. NEI Containment- Challenge 3 – Core exit thermocouples in excess of 1200 degrees and restoration procedures not effective within 15 minutes; or, core exit thermocouples in excess of 700 degrees with reactor vessel level below top of active fuel and restoration procedures not effective within 15 minutes
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. CSP-C.1, Response to Inadequate Core Cooling

6. **CET readings ≥700°F with RVLIS NR ≤25 ft and no RCPs running (Core Cooling-RED path, CSP-C.1) AND Restoration procedures not effective within 15 min.**

This threshold indicates significant core exit superheating and core uncovery. It must be assumed that the loss of RCS inventory is a result of a loss of the RCS barrier. These conditions, if not mitigated, will likely lead to core melt which will in turn result in a challenge of containment.

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 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 not been, or will not be effective.

For the purpose of this threshold the term 'effective' with regards to functional restoration procedures means that the specified criterion no longer exists.

Reference(s):

1. NEI Containment- Challenge 3 – Core exit thermocouples in excess of 1200 degrees and restoration procedures not effective within 15 minutes; or, core exit thermocouples in excess of 700 degrees with reactor vessel level below top of active fuel and restoration procedures not effective within 15 minutes
2. CSP-ST.0 Unit 1(2) Critical Safety Function Status Trees, Figure 2
3. CSP-C.1, Response to Inadequate Core Cooling

7. **Containment radiation ≥15,900 R/hr**

The containment radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Cladding barrier. NUREG-1228 "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents" states that such readings do not exist when the amount of cladding damage is less than 20%. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure into the reactor coolant has occurred. Regardless of whether the Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a challenge of the Containment barrier. The reading is higher than that specified for Fuel Cladding barrier Loss #4 and RCS barrier Loss #3. Containment radiation readings at or above the Containment barrier challenge threshold, therefore, signify a loss of two fission product barriers and challenge of a third, indicating the need to upgrade the emergency classification to a General Emergency.
It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping. Monitors used for this fission product barrier loss threshold are the containment high-range area monitors:

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

Reference(s):

1. NEI Containment- Challenge 6 – Containment rad monitor reading GREATER THAN (site-specific) R/hr
2. PBF 1608, Calculation 2004-0006
3. SAMG SAG-5, Reduce Fission Product Releases, Attachment D

8. **Emergency Director Judgment**

Emergency Director judgment addresses any other factors that are to be used in determining whether the containment is potentially lost. In addition, the inability to monitor containment integrity should also be considered in this threshold as a factor in judging that the containment may be considered potentially lost.

Reference(s):

1. NEI Containment- Challenge 8 – Emergency Director Judgment

Containment Loss

1. **Rapid unexplained containment pressure drop following initial rise**

   Rapid unexplained loss of pressure (i.e., not attributable to containment spray operation, running containment accident cooling units or condensation effects) following an initial pressure rise indicates a loss of both RCS and containment integrity. FSAR Figure 14.3.2-1 illustrates containment pressure response for a bounding LOCA. Containment pressure peaks at approximately 34 psia.

   Reference(s):

   1. NEI Containment- Loss 2 – Containment pressure
   2. FSAR Figure 14.3.2-1
   3. FSAR Tables 14.3.2-1 through 14.3.2-3
   4. FSAR 14.3.1, Small Break Loss of Coolant Accident Analysis
   5. FSAR 14.3.2, Large Break Loss-Of-Coolant Accident Analysis

2. **Containment pressure or sump level response not consistent with LOCA conditions**

   This threshold addresses unexpected changes occurring in containment pressure or sump level that are not explainable due to operator actions or automatic system actions. 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 containment pressure not rising indicates containment bypass and 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.

   Reference(s):

   1. NEI Containment- Loss 2 – Containment pressure
   2. FSAR Figure 14.3.2-1
   3. FSAR Tables 14.3.2-1 through 14.3.2-3
   4. FSAR 14.3.1, Small Break Loss of Coolant Accident Analysis
   5. FSAR 14.3.2, Large Break Loss-Of-Coolant Accident Analysis

3. **Ruptured S/G is also faulted outside of containment**

This "loss" threshold recognizes that S/G tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The "loss" threshold addresses the condition in which a ruptured steam generator (S/G) is also faulted. This condition represents a bypass of the RCS and Containment barriers. In conjunction with RCS barrier Loss #2, this would always result in the declaration of a Site 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 inclusion of thresholds that use Emergency Procedure terms like "ruptured" and "faulted" facilitates the classification process.

Reference(s):

1. NEI Containment- Loss 4 – SG Secondary Side Release with P-to-S Leakage
2. EOP-0, Reactor Trip or Safety Injection

4. **Primary-to-secondary leakage ≥10 gpm with non-isolable steam release from affected S/G to the environment**

This "loss" threshold recognizes that S/G tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. This condition represents a bypass of the RCS and Containment barriers. In conjunction with RCS barrier Loss #2, this would always result in the declaration of a Site Emergency.

The "loss" threshold addresses S/G tube leaks that exceed 10 gpm in conjunction with a non-isolable release path to the environment from the affected steam generator. The threshold for establishing the non-isolable secondary side release is intended to be a prolonged release of radioactivity from the affected 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 non-isolable release path to the environment. These minor releases are assessed using radiological effluent EAL thresholds.

A pressure boundary leakage of 10 gpm is also used as the threshold in EAL MU5.1, RCS Leakage. For smaller breaks, not exceeding the normal charging capacity threshold in RCS barrier Challenge #3 or not resulting in ECCS actuation in RCS barrier Loss #2, this threshold results in the declaration of an Unusual Event. For larger breaks, RCS barrier Challenge #3 and RCS barrier Loss #2 would result in an Alert. For S/G tube ruptures (SGTRs) which may involve more than one steam generator or unisolable secondary line breaks, this threshold would occur in conjunction with RCS barrier Loss #2 and would result in a Site Emergency. Escalation to General Emergency would be based on the challenge of the Fuel Cladding barrier.

There is some redundancy in the Containment loss thresholds #3 and #4. This was recognized during the development process. The inclusion of thresholds that use Emergency Procedure terms like "ruptured" and "faulted" facilitates the classification process.

Reference(s):

1. NEI Containment- Loss 4 – SG Secondary Side Release with P-to-S Leakage
2. EOP-0, Reactor Trip or Safety Injection

5. **Containment isolation required and containment isolation or ventilation valve(s) not closed when required**

AND

Radiological release pathway to the environment exists

This threshold addresses incomplete containment isolation that allows direct release to the environment. It represents a loss of both the RCS and Containment barriers and therefore warrants declaration of a Site Emergency. Failure of containment isolation or containment ventilation isolation valves to isolate when required addresses incomplete containment isolation that allows direct release to the environment. It represents a loss of both the RCS and Containment barriers.

Reference(s):

1. NEI Containment- Loss 5 – CNMT Isolation Valves Status After CNMT Isolation
2. CSP-Z.1, Attachment B, Containment Isolation Valves

6. **Inability to isolate any primary system discharging outside containment AND Radiological release pathway to the environment exists**

This threshold addresses primary systems (either direct or indirect) that are not considered in Containment loss #5. If the primary system cannot be isolated, a loss of both the RCS and the Containment barriers results. No leakage threshold is specified since leaks outside containment, particularly under dynamic conditions, are difficult to quantify and may manifest themselves with diverse symptoms. Symptoms of a primary system discharging outside containment may be indicated via mass balance, lowering RCS inventory without corresponding containment response, or area temperatures and radiation levels outside containment. It is for this reason that Emergency Director judgment should be used in evaluating this criterion. However, it is intended that the magnitude of the leak associated with this EAL be consistent with RCS barrier Challenge #3 of 60 gpm or greater.

Inability to isolate means that the leak cannot be isolated from the main control board.

Reference(s):

1. NEI Containment- Loss 5 – CNMT Isolation Valves Status After CNMT Isolation
2. CSP-Z.1, Attachment B, Containment Isolation Valves

7. **Emergency Director Judgment**

Emergency Director judgment addresses any other factors that are to be used in determining whether the Containment barrier is lost. The inability to monitor the containment integrity should also be considered in this threshold as a factor in judging that the Containment barrier may be considered lost.

Reference(s):

1. NEI Containment- Loss 8 – Emergency Director Judgment